NATIONAL ENRICHMENT FACILITY

SAFETY ANALYSIS REPORT

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ACRONYMS and ABBREVIATIONS

AC	alternating current
ACI	American Concrete Institute
ADEM	Alabama Department of Environmental Management
AEA	Atomic Energy Act
AEP	American Electric Power
AEGL	Acute Exposure Guideline Level
AHU	air handling unit
AISC	American Institute of Steel Construction
ALARA	as low as reasonably achievable
ALI	Annual Limit on Intake
ANPR	Advance Notice of Proposed Rulemaking
ANS	American Nuclear Society
ANSI	American National Standards Institute
AP	air particulate
APE	area of potential effects
AQB	Air Quality Bureau
ASCE	American Society of Civil Engineers
ASLB	Atomic Safety and Licensing Board
ASME	American Society of Mechanical Engineers
ASNT	American Society of Nondestructive Testing
ASTM	American Society for Testing Materials
ATSDR	Agency for Toxic Substances and Disease Registry
AVLIS	Atomic Vapor Laser Isotope Separation
BDC	baseline design criteria
BEA	Bureau of Economic Analysis
BLM	Bureau of Land Management
BMP	Best Management Practices
BNFL	British Nuclear Fuels
BNFL-EL	British Nuclear Fuels – Enrichment Limited
BOD	biochemical oxygen demand
BS	Bachelor of Science
CA	Controlled Area
CAA	Clean Air Act
CAAS	Criticality Accident Alarm System
CAB	Centrifuge Assembly Building
CAM	Continuous Air Monitor
CAP	Corrective Action Program
CBG	Census Block Group
CEDE	Committed Effective Dose Equivalent
CEQ	Council on Environmental Quality
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFO	Chief Financial Officer
CFR	Code of Federal Regulations
CHP	certified health physicist
CIS	Commonwealth of Independent States
CM	configuration management

ACRONYMS and ABBREVIATIONS

COD	chemical oxygen demand
COO	Chief Operating Officer
CRDB	Cylinder Receipt and Dispatch Building
CUB	Central Utilities Building
CVRF	Central Volume Reduction Facility
CWA	Clean Water Act
D&D	decontamination and decommissioning
DAC	derived air concentration
DBA	design basis accident
DBE	design basis earthquake
DCF	dose conversion factor
DE	Dose Equivalent
DEIS	Draft Environmental Impact Statement
DI	deionized
DOC	United States Department of Commerce
DOE	United States Department of Energy
DOI	United States Department of Interior
DOT	United States Department of Transportation
E	east
EDE	Effective Dose Equivalent
EECP	Entry/Exit Control Point
EIA	Energy Information Administration
EIS	Environmental Impact Statement
EJ	Environmental Justice
EMS	Emergency Medical Services
EOC	Emergency Operations Center
EPA	United States Environmental Protection Agency
EPCRA	Emergency Planning and Community Right-to-Know Act
EPRI	Electric Power Research Institute
eqs.	equations
ER	Environmental Report
ERPG	Emergency Response Planning Guideline
ENE	east north east
ESE	east south east
ETTP	East Tennessee Technology Park
FEIS	Final Environmental Impact Statement
FEMA	Federal Emergency Management Agency
FHA	fire hazards analysis
FNMC	Fundamental Nuclear Material Control
FR	Federal Register
FWPCA	Federal Water Pollution Control Act
GDP	Gaseous Diffusion Plant
GET	General Employee Training
GEVS	Gaseous Effluent Vent System
GPS	Global Positioning System
HEPA	high efficiency particulate air
HEU	highly enriched uranium
HMTA	Hazardous Materials Transportation Act
HS&E	Health, Safety, and Environment

ACRONYMS and ABBREVIATIONS

HUD	United States Department of Housing and Urban Development
HVAC	heating, ventilating, and air conditioning
HWA	Hazardous Waste Act
HWB	Hazardous Waste Bureau
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
INFL	International Nuclear Fuels Plc
I/O or I-O	input/output
IPD	Implicit Price Deflator
IROFS	items relied on for safety
ISA	Integrated Safety Analysis
ISO	International Organization for Standardization
JCIDA	Jackson County Industrial Development Authority
LAN	local area network
LCC	local control center
LCD	local climatic data
L_{dn}	Day-Night Average Sound Level
L_{eq}	Equivalent Sound Level
LES	Louisiana Energy Services
LEU	low enriched uranium
LLC	Limited Liability Company
LLD	lower limits of detection
LLNL	Lawrence Livermore National Laboratory
LLW	low-level waste
LOI	local operator interface
LQ	Location Quotients
LTA	lost time accident
LTC	load tap changer
LTTS	Low Temperature Take-off Station
M&TE	measuring and test equipment
MAPEP	Mixed Analyte Performance Evaluation Program
max.	maximum
MC&A	material control and accountability
MCL	maximum contaminant level
MCNP	Monte Carlo N-Particle
MDA	minimum detectable activity
MDC	minimum detectable concentration
ME&I	mechanical, electrical and instrumentation
min.	minimum
MM	modified mercalli
MMI	modified mercalli intensity
MOU	Memorandum of Understanding
MOX	mixed oxide fuel
MUA	multi-attribute utility analysis
N	north
NAAQS	National Ambient Air Quality Standards
NASA	National Aeronautic Space Administration
NCA	Noise Control Act
NCRP	National Council on Radiological Protection and Measurements

ACRONYMS and ABBREVIATIONS

NCS	nuclear criticality safety
NCSE	nuclear criticality safety evaluation
NDA	Non-destructive assessment
NE	Northeast
NEF	National Enrichment Facility
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NESHAPS	National Emission Standards for Hazardous Air Pollutants
NFPA	National Fire Protection Association
NHPA	National Historic Preservation Act
NELAC	National Environmental Laboratory Accreditation Conference
NIOSH	National Institute of Occupational Safety and Health
NIST	National Institute of Standards and Technology
NM	New Mexico
NMAC	New Mexico Administrative Code
NMDGF	New Mexico Department of Game and Fish
NMED	New Mexico Environmental Department
NMHWB	New Mexico Hazardous Waste Bureau
NMRPR	New Mexico Radiation Protection Regulations
NMSA	New Mexico State Agency
NMSE	New Mexico State Engineer
NMSHPO	New Mexico State Historic Preservation Office
NMSLO	New Mexico State Land Office
NMSS	Nuclear Material Safety and Safeguards
NMWQB	New Mexico Water Quality Bureau
NMWQCC	New Mexico Quality Control Commission
NNE	north-northeast
NNW	north-northwest
No.	number
NOAA	National Oceanic and Atmospheric Administration
NOI	Notice of Intent
NPDES	National Pollutant Discharge Elimination System
NPDWS	National Primary Drinking Water Standard
NRC	United States Nuclear Regulatory Commission
NRHP	National Register of Historic Places
NSDWS	National Secondary Drinking Water Standard
NSPS	New Source Performance Standards
NSR	New Source Review
NTS	Nevada Test Site
NWS	National Weather Service
NW	northwest
OEPA	Ohio Environmental Protection Agency
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
OVEC	Ohio Valley Electric Corporation
P&IDs	piping and instrumentation diagrams
p.	page
PA	public address
PEL	Permissible Exposure Level

ACRONYMS and ABBREVIATIONS

PFPE	perfluorinated polyether
PGA	peak ground acceleration
pH	measure of the acidity or alkalinity
PHA	Process Hazard Analysis
Ph.D.	Doctor of Philosophy
PIA	Potentially Impacted Area
PLC	Programmable Logic Controllers
PM	preventive maintenance
$PM_{2.5}$	particulates $\leq 2.5\mu m$
PM_{10}	particulates $\leq 10\mu m$
PMF	probable maximum flood
PMP	Probable Maximum Precipitation
PMWP	Probable Maximum Winter Precipitation
PORTS	Portsmouth Gaseous Diffusion Plant
POTW	Publicly Owned Treatment Works
pp.	pages
PRC	Peoples Republic of China
PSAR	Preliminary Safety Analysis Report
PSP	Physical Security Plan
QA	quality assurance
QAPD	Quality Assurance Program Description
QC	Quality Control
RCB	Radiation Control Bureau
RCRA	Resource Conservation and Recovery Act
RCZ	radiation control zone
REIS	Regional Economic Information System
REMP	Radiological Environmental Monitoring Program
RIMS	Regional Input-Output Modeling System
ROI	Region of Interest or Radius of Influence
RTE	Rare Threatened and Endangered
RWP	radiation work permit
S	south
SAR	Safety Analysis Report
SB	Separations Building
Sc.D.	Doctor of Science
SCRAM	Support Center for Regulatory Air Models
SDWA	Safe Drinking Water Act
SE	southeast
SER	Safety Evaluation Report
SHPO	State Historic Preservation Officer
SILEX	Separation of Isotopes by Laser Excitation
SNM	special nuclear material
SPCC	spill prevention, control, and countermeasures
SPL	Sound Level Pressure
SRC	Safety Review Committee
SSC	structure, system, and component
SSE	safe shutdown earthquake
SSE	south-southeast
SSW	south-southwest

ACRONYMS and ABBREVIATIONS

STEL	short term exposure limits
STP	standard temperature and pressure
SVOC	semivolatile organic compounds
SW	southwest
SWPPP	Storm Water Pollution Prevention Plan
TDEC	Tennessee Department of Environment and Conservation
TDS	Total Dissolved Solids
TEDE	total effective dose equivalent
TLD	thermoluminescent dosimeter
TN	Tennessee
TSB	Technical Services Building
TSP	total suspended particulates
TVA	Tennessee Valley Authority
TWA	time weighted average
TWDB	Texas Water Development Board
TX	Texas
UBC	Uranium byproduct cylinder
UCL	Urenco Capenhurst Limited
UCN	Ultra-Centrifuge Netherlands NV
UNAMAP	Users Network for Applied Modeling of Air Pollution
UPS	uninterruptible power supply
US	United States
USACE	United States Army Corps of Engineers
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
USDA	United States Department of Agriculture
USFWS	United States Fish and Wildlife Service
USGS	United States Geological Survey
UV	ultravoilet
VOC	volatile organic compound
W	West
WCS	Waste Control Specialists
WIPP	Waste Isolation Pilot Plant
WMA	wildlife management area
WNA	World Nuclear Association
WNW	west-northwest
WQB	Water Quality Bureau
WQCC	Water Quality Control Commission
WSW	west-southwest

UNITS OF MEASURE

Bq	Becquerel
BTU	british thermal unit
°C	degrees celsius
Ci	curie
cm	centimeter
d	day
dB	decibel
dBA	decibel A-weighted
dpm	disintegrations per minute
°F	degrees farenheit
ft	feet
g	gram
g_a	gravitational acceleration
gal	gallon
gpm	gallons per minute
Gy	Gray
ha	hectares
hp	horsepower
hr	hour
Hz	*hertz (cycle per second)*
in	inch
in. H_2O	inches of water (column)
J	Joule
kg	kilogram
km	kilometer
kWh	kilowatt-hour
L	liter
lb	pound
lbs	pounds
m	meter
mbar abs	millibar absolute
mbarg	millibar gauge
MBq	megabecquerel
mi	mile
min	minute
M_N	local magnitude
Mo	month
msl	mean sea level
MT or t	metric ton
MTU	Metric ton uranium
oz	ounce
Pa	pascal
ppb	parts per billion
ppm	parts per million
psia	pounds per square inch absolute
psig	pounds per square inch gauge
R	Roentgen
rad	radiation absorbed dose
rem	Roentgen equivalent man

UNITS OF MEASURE

scfm	standard cubic feet per minute
s	second
Sv	sievert
SWU	separative work unit
μmhos	micromhos
V	volt
VA	volt-ampere
W	watt
$^{w}/_{o}$	weight percent
χ/Q	atmospheric concentration per unit source
yd	yard
yr	year
σ	standard deviation

Pico (p)	X 10^{-12}
Nano (n)	X 10^{-9}
Micro (μ)	X 10^{-6}
Milli (m)	X 10^{-3}
Centi (c)	X 10^{-2}
Kilo (k)	X 10^{3}
Mega (M)	X 10^{6}

ECIFIED (BASIC SYMBOL FLANGED)

ITH BELLOWS

VE

VE

(ARROW INDICATED CONTROL
DED TO ANY VALVE TYPE)

PRESSURE

W LEFT TO RIGHT)

PE)

EN

PR101E	ROTARY VALVE
PR233	3-WAY MID PORT CLOSED (MIDCLOSE)
PR233A	3-WAY MID PORT CLOSED (SIDECLOSE)
PR234	CAMFLEX VALVE
PR235	TRIPLE DUTY VALVE
PR236	BALANCE/CIRCUIT SETTER VALVE
PR236	HIGH PURITY UPSTREAM PURGE POINTS
PR237A	HIGH PURITY DOWNSTREAM PURGE POINTS
PR237B	HIGH PURITY UPSTREAM AND DOWNSTREAM PURGE POINTS

PIPING LINE FEATURES & GENERAL EQUIPMENT

1.2.1	CONCENTRIC REDUCER
1.2.2	ECCENTRIC REDUCER (FLUSH TOP)
1.2.3	ECCENTRIC REDUCER (FLUSH BOTTOM)
1.2.4	FLEXIBLE PIPE OR BELLOWS (FLANGED)
1.2.5	SPRAY
1.2.6	SPRAY BAR
1.2.7	SIGHT FLOW INDICATOR
1.2.8	SIPHON DRAIN
1.2.9	VENT TO ATMOSPHERE
1.2.11	STRAINER OR FILTER (BASIC SYMBOL)
1.2.11	STRAINER 'Y' TYPE (FLANGED)
1.2.12	STRAINER BUCKET TYPE (FLANGED)
1.2.13	TRAP DRAIN (eg. CONDENSATE RELEASE)
1.2.14	TRAP VENT (eg. AUTOMATIC AIR VENT)
1.2.15	DRAIN
1.2.16	BURSTING DISC (FLANGE AND PIPE MAY BE ADDED TO OUTLET IF REQUIRED)
1.2.17	WEIGHING DEVICE (INCLUDING LOAD CELLS)
1.2.18	STACK
1.2.19	FLANGE
1.2.21	BLANK FLANGE
1.2.21	KLEIN COUPLING (KF FLANGE)
1.2.23	VACUUM FLANGE UCL ONLY (WITHOUT TEST CONNECTION)
1.2.24	ORIFICE PLATE
1.2.25	VENTURI
1.2.26	SCREWED END CAP
1.2.27	WELDED END CAP
1.2.28	HOSE CONNECTOR
1.2.29	QUICK RELEASE COUPLING
1.2.30	POINT OF CHANGE OF MATERIAL OR SYSTEM RESPONSIBILITY
1.2.31	AREA OR PACKAGE BOUNDARY
1.2.32	ARROW FOR INLET OR OUTLET AT CONTINUATION INTERFACE
1.2.33	INTERFACE OF QUALITY REQUIREMENTS (QS) UD ONLY
1.2.34	INTERFACE OF SUB SYSTEM (F1) UD ONLY
1.2.35	INTERFACE OF FUNCTION UNIT (F3) UD ONLY
1.2.36	INTERFACE OF COMPONENT UD ONLY
1.2.37_LGE	LINE CONTINUATION
PR032	TEMPORARY STRAINER
PR033	GENERIC COMPONENT
PR034	FLAME ARRESTER
PR035	REMOVABLE SPOOL PIECE
PR053	ROTATING SPRAY BALL
PR054	FIXED SPRAY BALL
PR056B	SIGHT GLASS LIGHT
PR058	EXPANSION JOINT
PR100	DUST COLLECTION HOSE
PR104	START-UP STRAINER
PR063	SIMPLEX STRAINER
PR064	DUPLEX STRAINER
PR066	Y STRAINER WITH VALVE

PR155

PR157

PR158

PR241

PR108

PR142

PR141

PR143

PR211

PR195

PR131A

PR131B

PIPE LINES

1.1.1

1.1.2

1.1.3

1.1.4

1.1.5

1.1.6

1.1.7

1.1.8

1.1.9

1.1.11

3LE COVER

VESSEL

T

R

1.5.2 HEAT EXCHANGER (BASIC SYMBOL)

1.5.3 HEATING / COOLING COIL (BASIC SYMBOL)

1.5.4 HEATING ELEMENT (el DENOTES ELECTRICAL)

1.5.5 VESSEL WITH EXTERNAL HEATING / COOLING COIL

1.5.6 VESSEL WITH INTERNAL HEATING / COOLING COIL

1.5.7 VESSEL WITH HEATING / COOLING JACKET

1.5.8 SHELL AND TUBE HEAT EXCHANGER

1.5.9 PLATE HEAT EXCHANGER

1.5.11 HEATING / COOLING COIL

1.5.11 FINNED TUBE HEAT EXCHANGER

1.5.12 COOLING TOWER FORCED DRAUGHT FANS INCLUDED AS APPROPRIATE (BASIC SYMBOL)

TRANSPORT EQUIPMENT

PR136 TANK CAR

PR137 TANK TRUCK

OTHER EQUIPMENT

PR198 PULSATION DAMPENER

PR111 EJECTOR

PR126 SPRAY DESUPERHEATER

PR139 STATIC MIXER

PR140 EDUCTOR MIXER

1.7.6_LGE FILTER PRESS

(BASIC SYMBOL)

1.7.2 COMPRESSOR (BASIC SYMBOL)

1.7.3 CENTRIFUGAL PUMP

3.1.1 VACUUM PUMP (TYPE NOT SPECIFIED)

3.1.2 SLIDING VANE ROTARY VACUUM PUMP

3.1.3 ROOTS VACUUM PUMP

3.1.4 HIGH PRESSURE COMPRESSOR

3.1.5 DIFFUSION PUMP

3.1.6 GETTER PUMP

PR102 CENTRIFUGAL (SIDE DISCHARGE)

PR103 ROTARY WITH PRESSURE RELIEF

PR104 SUMP

PR104A SUMP WITH SUBMERSIBLE MOTOR

PR105 METERING

1.7.4_LGE AIR DIAPHRAGM

PR134 SINGLE DIAPHRAGM

ROTARY

PR133 RECIPROCATION COMPRESSOR

PR107 VACUUM PUMP

PR135 PROGRESSIVE CAVITY

VACUUM EQUIPMENT

3.2.1 OIL TRAP

3.2.2 ADSORPTION TRAP (BASIC SYMBOL)

3.2.3 UF6 CYLINDER 30" AND 48"

3.2.4 SAMPLE BOTTLE

3.2.5 LIQUID NITROGEN DEWAR

3.2.6 FILTER CARTRIDGE HOLDER

3.2.7 FILTER CARTRIDGE HOLDER WITH FILTER

3.2.8 MONOBED FILTER

3.2.9 MIXED-BED FILTER

3.2.10 ABSORPTION FILTER (OIL COLLECTION)

3.2.11 COLD TRAP

2.1.11

2.1.11

2.1.12

2.1.13

2.1.14

2.1.15

2.1.1

2.1.2

2.1.3

2.1.4

2.1.5

2.1.6

2.1.7

2.1.8

1.7.5_LGE

NOZZLES

1.7.5_LGE C

PR101B T

PR101C Π

PR101D

OPERATED ACTUATING
CLOSES ON FAILURE
IERGY)
OPERATED ACTUATING
OPEN ON FAILURE
IERGY)
OPERATED ACTUATING
RETAINS POSITION
CTUATING ENERGY)
ATOR
REDUCING REGULATOR
)
D ACTUATING ELEMENT
JATOR
JATOR
JATOR
TOR
TOR
TOR
LATOR VALVE
PRESSURE TAP)
UATOR WITH FLOAT
RESSURE REGULATOR
D
ECTION ARROW
D
OR
RESSURE ACTUATOR
OR
R ON PNEUMATIC
ULIC ACTUATOR
TOR

PR003	TURBINE OR PROPELLER TYPE PRIMARY ELEMENT
PR004	PITOT TYPE SENSOR
PR005	VORTEX SENSOR
PR013	RESTRICTION ORIFICE (FLANGED)
PR014	ORIFICE UNION (SCREWED)
PR174	QUICK CHANGE ORIFICE INSTRUMENT
PR174	SINGLE PORT PITOT TUBE INSTRUMENT
PR176	DOUBLE PORT PITOT TUBE INSTRUMENT
PR177	DIAPHRAGM SEAL
PR208	PIG TAIL
PR192	THERMOWELL
PR192	FLUME INSTRUMENT
PR179	WEIR INSTRUMENT
PR180	FLOW VANE INSTRUMENT
PR006	PRESSURE RELIEF RUPTURE DISK
PR006A	VACUUM RELIEF RUPTURE DISK
PR007	CHEMICAL SEAL
PR131A	PRESSURE RELIEF VALVE
PR131B	ANGLE VACUUM RELIEF VALVE
PR131C	PRESSURE & VACUUM RELIEF VALVE
PR011	CONSERVATION VENT (PRESSURE SAFETY VACUUM)
PR181	VARIABLE FLOW INSTRUMENT (ROTAMETER)

PSV

XX

4.1.2	DISPLAYED IN LOCAL PANEL
4.1.3	DISPLAYED IN CONTROL ROOM
4.2.1	DISPLAYED ON EQUIPMENT OR IN PROCESS LINE (VALVE)
4.2.2	DISPLAYED IN LOCAL PANEL (VALVE)
4.2.3	DISPLAYED IN CONTROL ROOM (VALVE)
4.1.4_LGE	PILOT LIGHT FIELD MOUNTED
4.1.5_LGE	GROUP CONTROL - REAR OF CONTROL ROOM PANEL

	FIRST LETTER	
	MEASURED OR INITIATING VARIABLE	MODI
A		
B		
C		
D	DENSITY	DIFFEF
E	ALL ELECTRICAL VARIAIBLES	
F	FLOW RATE	RA
G	GAUGING, POSITION, OR LENGTH	
H	HAND (MANUALLY INITIATED) OPERATED	
I		
J		SC
K	TIME ORTIME PROGRAM	
L	LEVEL	
M	MOISTURE OR HUMIDITY	
N	USER'S CHOICE	
O	USER'S CHOICE	
P	PRESSURE OR VACUUM	
Q	QUANTITY FOR EXAMPLE ANALYSIS, CONCENTRATION, CONDUCTIVITY	INTEGRATE (
R	NUCLEAR RADIATION	
S	SPEED OR FREQUENCY	
T	TEMPERATURE	
U	MULTIVARIABLE	
V	VISCOSITY	
W	WEIGHT OR FORCE	
X	USER'S CHOICE	
Y	USER'S CHOICE	
Z		
α		ALPHA
β		BETA
γ		GAMMA

ALL LETTER CODES SHALL BE IN UPPER CASE W
MODIFIERS IN COLUMN 3, THESE SHALL BE IN LO

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1.0 GENERAL INFORMATION

This section contains a general description and purpose of the Louisiana Energy Services (LES) National Enrichment Facility (NEF). The facility enriches uranium for producing nuclear fuel for use in commercial power plants. This Safety Analysis Report (SAR) follows the format recommended by NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility (NRC, 2002). The level of detail provided in this chapter is appropriate for general familiarization and understanding of the facility and processes. The information is to be used as background for the more detailed descriptions provided in other chapters of the license application. Cross-references to the more detailed descriptions are provided in this chapter. This chapter also provides information on the corporate structure and economic qualifications of LES.

1.1 FACILITY AND PROCESS DESCRIPTION

The NEF, a state–of–the–art process plant, is located in southeastern New Mexico in Lea County approximately 0.8 km (0.5 mi) west of the Texas state border. This location is approximately 8 km (5 mi) due east of Eunice and 32 km (20 mi) south of Hobbs.

The geographic location of the facility is shown on Figures 1.1-1, State Map, and 1.1-2, County Map.

This uranium enrichment plant is based on a highly reliable gas centrifuge process. The plant is designed to separate a feed stream containing the naturally occurring proportions of uranium isotopes into a product stream - enriched in the uranium-235 (^{235}U) isotope and a tails stream - depleted in the ^{235}U isotope. The process, entirely physical in nature, takes advantage of the tendency of materials of differing density to segregate in the force field produced by a centrifuge. The chemical form of the working material of the plant, uranium hexafluoride (UF_6), does not require chemical transformations at any stage of the process. This process enriches natural UF_6, containing approximately 0.711% ^{235}U to a UF_6 product, containing ^{235}U enriched up to 5 $^w/_o$.

The nominal capacity of the facility is 3 million separative work units (SWU) per year. The maximum gross output of the facility is slightly greater than 3 million SWU thus allowing for a production margin for centrifuge failures and occasional production losses during the operational lifetime of the facility.

Feed is received at the plant in specially designed cylinders containing up to 12.7 MT (14 tons) of UF_6. The cylinders are inspected and weighed in the Cylinder Receipt and Dispatch Building (CRDB) and transferred to the main process facility, the Separations Building. Separation operations are divided among three Separations Building Modules, each capable of handling approximately one-third of plant capacity. Each Separations Building Module is divided into two Cascade Halls, and each Cascade Hall is comprised of eight cascades. Therefore, the total plant is comprised of 48 cascades. Each Cascade Hall produces enriched UF_6 at a specified assay ($^w/_o$ ^{235}U), so up to six different assays can be produced at one time.

The enrichment process, housed in the Separations Building, is comprised of four major elements: a UF_6 Feed System, a Cascade System, a Product Take-off System, and a Tails Take-off System. Other product related functions include the Product Liquid Sampling and Product Blending Systems. Supporting functions include sample analysis, equipment decontamination and rebuild, liquid effluent treatment and solid waste management.

The major equipment used in the UF_6 feed process are Solid Feed Stations. Feed cylinders are loaded into Solid Feed Stations; vented for removal of light gases, primarily air and hydrogen fluoride (HF), and heated to sublime the UF_6. The light gases and UF_6 gas generated during feed purification are routed to the Feed Purification Subsystem where the UF_6 is desublimed.

The major pieces of equipment in the Feed Purification Subsystem are UF_6 Cold Traps, a Vacuum Pump/Chemical Trap Set, and a Low Temperature Take-off Station (LTTS). The Feed Purification Subsystem removes any light gases such as air and HF from the UF_6 prior to introduction into the cascades. The UF_6 is captured in UF_6 Cold Traps and ultimately recycled as feed, while HF is captured on chemical traps.

After purification, UF_6 from the Solid Feed Stations is routed to the Cascade System. Pressure in all process lines is subatmospheric.

Gaseous UF_6 from the Solid Feed Stations is routed to the centrifuge cascades. Each centrifuge has a thin-walled, vertical, cylindrically shaped rotor that spins around a central post within an outer casing. Feed, product, and tails streams enter and leave the centrifuge through the central post. Control valves, restrictor orifices, and controllers provide uniform flow of product and tails.

Depleted UF_6 exiting the cascades is transported from the high vacuum of the centrifuge for desublimation into Uranium Byproduct Cylinders (UBCs) at subatmospheric pressure. The primary equipment of the Tails Take-off System is the vacuum pumps and the Tails Low Temperature Take-off Stations (LTTS). Chilled air flows over cylinders in the Tails LTTS to effect the desublimation. Filling of the cylinders is monitored with a load cell system, and filled cylinders are transferred to an outdoor storage area (UBC Storage Pad).

Enriched UF_6 from the cascades is desublimed in a Product Take-off System comprised of vacuum pumps, Product Low Temperature Take-off Stations (LTTS), UF_6 Cold Traps, and Vacuum Pump/Chemical Trap Sets. The pumps transport the UF_6 from the cascades to the Product LTTS at subatmospheric pressure. The heat of desublimation of the UF_6 is removed by cooling air routed through the LTTS. The product stream normally contains small amounts of light gases that may have passed through the centrifuges. Therefore, a UF_6 Cold Trap and Vacuum Pump/Trap Set are provided to vent these gases from the product cylinder. Any UF_6 captured in the cold trap is periodically transferred to another product cylinder for use as product or blending stock. Filling of the product cylinders is monitored with a load cell system, and filled cylinders are transferred to the Product Liquid Sampling System for sampling.

Sampling is performed to verify product assay level ($^{w}/_{o}$ ^{235}U). The Product Liquid Sampling Autoclave is an electrically heated, closed pressure vessel used to liquefy the UF_6 and allow collection of a sample. The autoclave is fitted with a hydraulic tilting mechanism that elevates one end of the autoclave so that liquid UF_6 pours into a sampling manifold connected to the cylinder valve. After sampling, the autoclave is brought back to the horizontal position and the cylinder is indirectly cooled by water flowing through coils located on the outer shell of the autoclave.

LES customers may require product at enrichment levels other than that produced by a single Cascade Hall. Therefore, the plant has the capability to blend enriched UF_6 from two donor cylinders of different assays into a product receiver cylinder. The Product Blending System is comprised of Blending Donor Stations for the two donor cylinders and a Blending Receiver Station for the receiver cylinder. The Donor Stations are similar to the Solid Feed Stations described earlier. The Receiver Station is similar to the Low-Temperature Take-off Stations described earlier.

Support functions, including sample analysis, equipment decontamination and rebuild, liquid effluent treatment and solid waste management are conducted in the Technical Services Building (TSB). Decontamination, primarily of pumps and valves, uses solutions of citric acid. Sampling includes a Chemical Laboratory for verifying product UF_6 assay, and an Environmental Monitoring Laboratory. Liquid effluent is collected and treated and monitored before discharge to the Treated Effluent Evaporation Basin, a double-lined evaporative basin with leak detection.

1.1.1 Facility Location, Site Layout, And Surrounding Characteristics

Site features are well suited for the location of a uranium enrichment facility as evidenced by its favorable conditions of hydrology, geology, seismology and meteorology as well as good transportation routes for transporting feed and product by truck.

The facility is located on approximately 220 ha (543 acres) of land in Section 32 of Lea County, New Mexico. The Separations Building Modules, Administration Building, Cylinder Receipt and Dispatch Building, Centrifuge Assembly Building, Central Utilities Building, Technical Services Building, and UBC Storage Pad are located approximately in the center of the Section on 73 ha (180 acres) of developed area. A Plot Plan of the facility is shown in Figure 1.1-3, Plot Plan (1 Mile Radius). The Facility Layout (Site Plan) depicting the Site Boundary and Controlled Area Boundary is shown in Figure 1.1-4, Facility Layout (Site Plan) with Site Boundary and Controlled Access Area Boundary.

The site lies along the north side of New Mexico Highway 234. It is relatively flat with slight undulations in elevation ranging from 1,033 to 1,061 m (3,390 to 3,430 ft) above mean sea level (msl). The overall slope direction is to the southwest. A barbed wire fence runs along the east, south and west property lines. The fence along the north property line has been dismantled. A 254-mm (10-in) diameter, underground carbon dioxide pipeline owned by Trinity Pipeline LLC, traverses the site from southeast to northwest. A 406-mm (16-in) diameter, underground natural gas pipeline, owned by the Sid Richardson Energy Services Company, is located along the south property line, paralleling New Mexico Highway 234.

The nearest community is Eunice, approximately 8 km (5 mi) from the site. There are no residences, schools, stores or other population centers within a 1.6 km (1 mi) radius of the site.

Additional details of proximity to nearby populations are provided in the Environmental Report.

1.1.2 Facilities Description

The major structures and areas of the facility are outlined below.

Separations Building Modules

The overall layout of a Separations Building Module is presented in Figures 1.1-5 through 1.1-7 and the UF_6 Handling Area is shown in Figure 1.1-8, UF_6 Handling Area Equipment Location. The facility includes three identical Separations Building Modules. Each module consists of two Cascade Halls, each having eight cascades with each cascade having hundreds of centrifuges. Each Cascade Hall is capable of producing approximately 500,000 SWU per year. The major functional areas of the Separations Building Modules are:

- Cascade Halls (2)
- Process Services Area
- UF_6 Handling Area

Source material and special nuclear material (SNM) are used or produced in this area.

Technical Services Building

The overall layout of the Technical Services Building (TSB) is presented in Figures 1.1-9, Technical Services Building First Floor, and 1.1-10, Technical Services Building Second Floor. The TSB contains support areas for the facility. It also acts as the secure point of entry to the Separations Building Modules and the Cylinder Receipt and Dispatch Building (CRDB). The major functional areas of the TSB are:

- Solid Waste Collection Room
- Vacuum Pump Rebuild Workshop
- Decontamination Workshop
- Ventilated Room
- Cylinder Preparation Room
- Mechanical, Electrical and Instrumentation (ME&I) Workshop
- Liquid Effluent Collection and Treatment Room
- Laundry
- TSB Gaseous Effluent Vent System (GEVS) Room
- Mass Spectrometry Laboratory
- Chemical Laboratory
- Environmental Monitoring Laboratory
- Truck Bay/Shipping and Receiving Area
- Medical Room
- Radiation Monitoring Control Room
- Break Room
- Control Room
- Training Room
- Security Alarm Center

Source material and SNM are found in this area.

Centrifuge Assembly Building

This building is used to assemble centrifuges before they are moved into the Separations Building and installed in the cascades. The overall layout of the Centrifuge Assembly Building (CAB) is presented in Figures 1.1-11 through 1.1-13. The Centrifuge Assembly Building is located adjacent to the Cylinder Receipt and Dispatch Building. The major functional areas of the CAB are:

- Centrifuge Component Storage Area
- Centrifuge Assembly Area

- Assembled Centrifuge Storage Area
- Centrifuge Test Facility
- Centrifuge Post Mortem Facility

Source material and SNM are used and produced in this area.

Administration Building

The general office areas and Entrance Exit Control Point (EECP) are located in the Administration Building, Figure 1.1-14, Administration Building. All personnel access to the facility occurs at this location. Vehicular traffic passes through a security checkpoint before being allowed to park. Parking is located outside of the Controlled Access Area (CAA) security fence. Personnel enter the Administration Building and general office areas via the main lobby.

Personnel requiring access to facility areas or the CAA must pass through the EECP. The EECP is designed to facilitate and control the passage of authorized facility personnel and visitors.

Entry to the facility area from the Administration Building is only possible through the EECP.

Security Building

The main site Security Building is located at the entrance to the plant. It functions as a security checkpoint for incoming and outgoing vehicular traffic. Employees, visitors and trucks that have access approval are screened at this location.

A guard house is located at the secondary site entrance on the west side of the site. Common carriers, such as mail delivery trucks, are screened at this location.

Cylinder Receipt and Dispatch Building

The overall layout of the Cylinder Receipt and Dispatch Building (CRDB) is presented in Figures 1.1-15, Cylinder Receipt and Dispatch Building First Floor Part A, and 1.1-16, Cylinder Receipt and Dispatch Building First Floor Part B. The CRDB is located between two Separations Building Modules, adjacent to the Blending and Liquid Sampling Area. This building contains equipment to receive, inspect, weigh and temporarily store cylinders of feed UF_6 sent to the plant; temporarily store, inspect, weigh, and ship cylinders of enriched UF_6 to facility customers; receive, inspect, weigh, and temporarily store clean empty product and UBCs prior to being filled in the Separations Building; and inspect, weigh, and transfer filled UBCs to the UBC Storage Pad. The functions of the Cylinder Receipt and Dispatch Building are:

- Loading and unloading of cylinders
- Inventory weighing
- Storage of protective cylinder overpacks
- Storage of clean empty and empty UBCs
- Buffer storage of feed cylinders

Source and SNM are used in this area.

Blending and Liquid Sampling Area

The Blending and Liquid Sampling Area is adjacent to the CRDB and is located between two Separations Building Modules. The Blending and Liquid Sampling Area is shown in Figure 1.1-17, Blending and Liquid Sampling Area First Floor.

The primary function of the Blending and Liquid Sampling Area is to provide means to fill ANSI N14.1 (ANSI, applicable version) Model 30B cylinders with UF_6 at a required ^{235}U enrichment level and to liquefy, homogenize and sample 30B cylinders prior to shipment to the customer. The area contains the major components associated with the Product Liquid Sampling System and the Product Blending System.

SNM is used in this area.

UBC Storage Pad

The facility utilizes an area outside of the CRDB, the UBC Storage Pad, for storage of cylinders containing UF_6 that is depleted in ^{235}U. The cylinder contents are stored under vacuum in corrosion-resistant ANSI N14.1 (ANSI, applicable version) Model 48Y cylinders.

The UBC storage area layout is designed for moving the cylinders with a small truck and a crane. A flatbed truck moves the UBCs from the CRDB to the UBC Storage Pad entrance. A double girder gantry crane removes the cylinders from the flatbed truck and places them in the UBC Storage Pad. The gantry crane is designed to double stack the cylinders in the storage area.

Source material is used in this area.

Central Utilities Building

The Central Utilities Building (CUB) is shown on Figure 1.1-18, Central Utilities Building. The Central Utilities Building houses two diesel generators, which provide the site with standby power. The rooms housing the diesel generators are constructed independent of each other with adequate provisions made for maintenance, equipment removal and equipment replacement, by including roll-up access doors. The building also contains Electrical Rooms, an Air Compressor Room, a Boiler Room and Cooling Water Facility.

Visitor Center

A Visitor Center is located outside of the Controlled Access area.

1.1.3 Process Descriptions

This section provides a description of the various processes analyzed as part of the Integrated Safety Analysis. A brief overview of the entire enrichment process is provided followed by an overview of each major process system.

1.1.3.1 Process Overview

The enrichment process at the NEF is basically the same process described in the SAR for the Claiborne Enrichment Center (LES, 1991). The Nuclear Regulatory Commission (NRC) staff documented its review of the Claiborne Enrichment Center license application and concluded that LES's application provided an adequate basis for safety review of facility operations and that construction and operation of the Claiborne Enrichment Center would not pose an undue risk to public health and safety (NRC, 1993). The design of the NEF incorporates the latest safety improvements and design enhancements from the Urenco enrichment facilities currently operating in Europe.

The primary function of the facility is to enrich natural uranium hexafluoride (UF_6) by separating a feed stream containing the naturally occurring proportions of uranium isotopes into a product stream enriched in ^{235}U and a tails stream depleted in the ^{235}U isotope. The feed material for the enrichment process is uranium hexafluoride (UF_6) with a natural composition of isotopes ^{234}U, ^{235}U, and ^{238}U. The enrichment process is a mechanical separation of isotopes using a fast rotating cylinder (centrifuge) based on a difference in centrifugal forces due to differences in molecular weight of the uranic isotopes. No chemical changes or nuclear reactions take place. The feed, product, and tails streams are all in the form of UF_6.

1.1.3.2 Process System Descriptions

An overview of the four enrichment process systems and the two enrichment support systems is discussed below.

Numerous substances associated with the enrichment process could pose hazards if they were released into the environment. Chapter 6, Chemical Process Safety, contains a discussion of the criteria and identification of the chemicals of concern at the NEF and concludes that uranium hexafluoride (UF_6) is the only chemical of concern that will be used at the facility. Chapter 6, Chemical Process Safety, also identifies the locations where UF_6 is stored or used in the facility and includes a detailed discussion and description of the hazardous characteristics of UF_6 as well as a detailed listing of other chemicals that are in use at the facility.

The enrichment process is comprised of the following major systems:

<u>UF_6 Feed System</u>

The first step in the process is the receipt of the feed cylinders and preparation to feed the UF_6 through the enrichment process.

Natural UF_6 feed is received at the NEF in 48Y or 48X cylinders from a conversion plant. Pressure in the feed cylinders is below atmospheric (vacuum) and the UF_6 is in solid form.

The function of the UF_6 Feed System is to provide a continuous supply of gaseous UF_6 from the feed cylinders to the cascades. There are six Solid Feed Stations per Cascade Hall; three stations in operation and three on standby. The maximum feed flow rate is 187 kg/hr (412 lb/hr) UF_6 based on a maximum capacity of 545,000 SWU per year per Cascade Hall.

Cascade System

The function of the Cascade System is to receive gaseous UF_6 from the UF_6 Feed System and enrich the ^{235}U isotope in the UF_6 to a maximum of 5 $^w/_o$.

Multiple gas centrifuges make up arrays called cascades. The cascades separate gaseous UF_6 feed with a natural uranium isotopic concentration into two process flow streams – product and tails. The product stream is the enriched UF_6 stream, from 2 - 5 $^w/_o$ ^{235}U, with an average of 4.5 $^w/_o$ ^{235}U. The tails stream is UF_6 that has been depleted of ^{235}U isotope to 0.20 – 0.34 $^w/_o$ ^{235}U, with an average of 0.32 $^w/_o$ ^{235}U.

Product Take-off System

The function of the Product Take-off System is to provide continuous withdrawal of the enriched gaseous UF_6 product from the cascades and to purge and dispose of light gas impurities from the enrichment process.

The product streams leaving the eight cascades are brought together into one common manifold from the Cascade Hall. The product stream is transported via a train of vacuum pumps to Product LTTS in the UF_6 Handling Area. There are five Product LTTS per Cascade Hall; two stations in operation and three stations on standby.

The Product Take-off System also contains a system to purge light gases (typically air and hydrogen fluoride) from the enrichment process. This system consists of UF_6 Cold Traps which capture UF_6 while leaving the light gas in a gaseous state. The cold trap is followed by product vent Vacuum Pump/Trap Sets, each consisting of a carbon trap, an alumina trap, and a vacuum pump. The carbon trap removes small traces of UF_6 and the alumina trap removes any hydrogen fluoride (HF) from the product gas.

Tails Take-off System

The primary function of the Tails Take-off System is to provide continuous withdrawal of the gaseous UF_6 tails from the cascades. A secondary function of this system is to provide a means for removal of UF_6 from the centrifuge cascades under abnormal conditions.

The tails stream exits each Cascade Hall via a primary header, goes through a pumping train, and then to Tails LTTS in the UF_6 Handling Area. There are ten Tails LTTS per Cascade Hall. Under normal operation, seven of the stations are in operation receiving tails and three are on standby.

In addition to the four primary systems listed above, there are two major support systems:

Product Blending System

The primary function of the Product Blending System is to provide a means to fill 30B cylinders with UF_6 at a specific enrichment of ^{235}U to meet customer requirements. This is accomplished by blending (mixing) UF_6 at two different enrichment levels to one specific enrichment level. The system can also be used to transfer product from a 30B or 48Y cylinder to another 30B cylinder without blending.

This system consists of Blending Donor Stations (which are similar to the Solid Feed Stations) and Blending Receiver Stations (which are similar to the Product LTTS) described under the primary systems.

Product Liquid Sampling System

The function of the Product Liquid Sampling System is to obtain an assay sample from filled product 30B cylinders. The sample is used to validate the exact enrichment level of UF_6 in the filled product cylinders before the cylinders are sent to the fuel processor.

This is the only system in the NEF that changes solid UF_6 to liquid UF_6.

1.1.4 Raw Materials, By-Products, Wastes, And Finished Products

The facility handles Special Nuclear Material of ^{235}U contained in uranium enriched above natural but less than or equal to 5.0 $^{w}/_{o}$ in the ^{235}U isotope. The ^{235}U is in the form of uranium hexafluoride (UF_6). The facility processes approximately 690 feed cylinders (Model 48Y or 48X), 350 product cylinders (Model 30B), and 625 UBCs (Model 48Y) per year.

LES does not propose possession of any reflectors or moderators with special characteristics.

Solid Waste Management

Solid waste generated at the NEF will be grouped into industrial (non-hazardous), radioactive, hazardous, and mixed waste categories. In addition, solid radioactive and mixed waste is further segregated according to the quantity of liquid that is not readily separable from the solid material. The solid waste management systems are comprised of a set of facilities, administrative procedures, and practices that provide for the collection, temporary storage, processing, and transportation for disposal of categorized solid waste in accordance with regulatory requirements. All solid radioactive wastes generated are Class A low-level wastes (LLW) as defined in 10 CFR 61 (CFR, 2003a).

Radioactive waste is collected in labeled containers in each Radiation Area and transferred to the Solid Waste Collection Room for processing. Suitable waste will be volume-reduced, and all radioactive waste will be disposed of at a licensed LLW disposal facility.

Hazardous waste and a small amount of mixed waste are generated at the NEF. These wastes are also collected at the point of generation and transferred to the Solid Waste Collection Room. Any mixed waste that may be processed to meet land disposal requirements may be treated in its original collection container and shipped as LLW for disposal.

Industrial waste, including miscellaneous trash, filters, resins and paper is shipped offsite for compaction and then sent to a licensed waste landfill.

Effluent Systems

The following NEF systems handle wastes and effluent.

- Separations Building Gaseous Effluent Vent System
- TSB Gaseous Effluent Vent System
- Liquid Effluent Collection and Treatment System
- Centrifuge Test and Post Mortem Facilities Exhaust Filtration System
- Septic System
- Solid Waste Collection System

- Decontamination System
- Fomblin Oil Recovery System
- Laundry System

Effluent Quantities

Quantities of radioactive and non-radioactive wastes and effluent are estimated and shown in the tables referenced in this section. The tables include quantities and average uranium concentrations. Portions of the waste considered hazardous or mixed are identified.

The following tables address plant effluents:

- Table 1.1-1, Estimated Annual Gaseous Effluent
- Table 1.1-2, Estimated Annual Radiological and Mixed Wastes
- Table 1.1-3, Estimated Annual Liquid Effluent
- Table 1.1-4, Estimated Annual Non-Radiological Wastes

Radioactive concentration limits and handling for liquid wastes and effluents are detailed in the Environmental Report.

The waste and effluent estimates described in the tables listed above were developed specifically for the NEF. Each system was analyzed to determine the wastes and effluents generated during operation. These values were analyzed and a waste disposal path was developed for each. LES considered the facility site, facility operation, applicable Urenco experience, applicable regulations, and the existing U.S. waste processing/disposal infrastructure during the development of the paths. The Liquid Effluent Collection and Treatment System and the Solid Waste Collection System were designed to meet these criteria.

Construction Wastes

During construction, efforts are made to minimize the environmental impact. Erosion, sedimentation, dust, smoke, noise, unsightly landscape, and waste disposal are controlled to practical levels and applicable regulatory limits. Wastes generated during site preparation and construction will be varied, depending on the activities in progress. The bulk of the wastes will consist of non-hazardous materials such as packing materials, paper and scrap lumber. These wastes will be transported off site to an approved landfill. It is estimated that the NEF will generate a non-compacted average waste volume of 3,058 m^3 (4,000 yd^3) annually.

Hazardous type wastes that may be generated during construction have been identified and annual quantities estimated are shown in Table 1.1-5, Annual Hazardous Construction Wastes. Any of these wastes that are generated will be handled by approved methods and shipped off site to approved disposal sites.

Management and disposal of all wastes from the NEF site will be performed by personnel trained to properly identify, store, and ship wastes, audit vendors, direct and conduct spill cleanup, provide interface with state agencies, maintain inventories and provide annual reports.

A Spill Prevention, Control and Countermeasure Plan (SPCC) will be implemented during construction to minimize the possibility of spills of hazardous substances, minimize environmental impact of any spills and ensure prompt and appropriate remediation. The SPCC plan will identify sources, locations and quantities of potential spills and response measures. The plan will identify individuals and their responsibilities for implementation of the plan and provide for prompt notifications of state and local authorities.

1.2 INSTITUTIONAL INFORMATION

This section addresses the details of the applicant's corporate identity and location, applicant's ownership organization and financial information, type, quarterly, and form of licensed material to be used at the facility, and the type(s) of license(s) being applied for.

1.2.1 Corporate Identity

1.2.1.1 Applicant

The Applicant's name, address, and principal office are as follows:

Louisiana Energy Services, L.P.
100 Sun Avenue NE, Suite 204
Albuquerque, NM 87109

The Applicant also maintains an office in Washington, DC during the licensing period at the following location:

2600 Virginia Avenue NW, Suite 610
Washington, D.C. 20037

1.2.1.2 Organization and Management of Applicant

Louisiana Energy Services (LES), L.P. is a Delaware limited partnership. It has been formed solely to provide uranium enrichment services for commercial nuclear power plants. LES has one, 100% owned subsidiary, operating as a limited liability company, formed for the purpose of purchasing Industrial Revenue Bonds and no divisions. The general partners are as follows:

A. Urenco Investments, Inc. (a Delaware corporation and wholly-owned subsidiary of Urenco Limited, a corporation formed under the laws of the United Kingdom ("Urenco") and owned in equal shares by BNFL Enrichment Limited ("BNFL-EL"), Ultra-Centrifuge Nederland NV ("UCN"), and Uranit GmbH ("Uranit") companies formed under English, Dutch and German law, respectively; BNFL-EL is wholly-owned by British Nuclear Fuels plc, which is wholly-owned by the Government of the United Kingdom; UCN is 99% owned by the Government of the Netherlands, with the remaining 1% owned collectively by the Royal Dutch Shell Group, DSM, Koninklijke Philips Electronics N.V. and Stork N.V.; Uranit is owned by Eon Kernkraft GmbH (50%) and RWE Power AG (50%), which are corporations formed under laws of the Federal Republic of Germany); and

B. Westinghouse Enrichment Company, LLC (a Delaware limited liability company and wholly-owned subsidiary of Westinghouse Electric Company LLC, a Delaware limited liability company ("Westinghouse"), whose ultimate parent, through two intermediary Delaware corporations and one corporation formed under the laws of the United Kingdom, is British Nuclear Fuels plc, which is wholly-owned by the Government of the United Kingdom).

The names and addresses of the responsible officials for the general partners are as follows:

Urenco Investments, Inc.
Charles W. Pryor, President and CEO
2600 Virginia Avenue NW, Suite 610
Washington, DC 20037

Dr. Pryor is a citizen of the United States of America

Westinghouse Enrichment Company, LLC
Ian B. Duncan, President
4350 Northern Pike
Monroeville, PA 15146

Mr. Duncan is a citizen of the United Kingdom.

The limited partners are as follows:

A. Urenco Deelnemingen B.V. (a Netherlands corporation and wholly-owned subsidiary of Urenco Nederlands B.V. (UNL);

B. Westinghouse Enrichment Company, LLC (the Delaware limited liability company, wholly-owned by Westinghouse, that also is acting as a General Partner);

C. Entergy Louisiana, Inc. (a Louisiana corporation and wholly-owned subsidiary of Entergy Corporation, a publicly-held Delaware corporation and a public utility holding company);

D. Claiborne Energy Services, Inc. (a Louisiana corporation and wholly-owned subsidiary of Duke Energy Corporation, a publicly-held North Carolina corporation);

E. Cenesco Company, LLC (a Delaware limited liability company and wholly-owned subsidiary of Exelon Generation Company, LLC, a Pennsylvania limited liability company);

F. Penesco Company, LLC (a Delaware limited liability company and wholly-owned subsidiary of Exelon Generation Company, LLC, a Pennsylvania limited liability company).

Urenco owns 70.5% of the partnership while Westinghouse owns 19.5% of LES. The remaining 10% is owned by the companies representing the three electric utilities, i.e., Entergy Corporation, Duke Energy Corporation, and Exelon Generation Company, LLC.

The President of LES is E. James Ferland, a citizen of the United States of America. LES' principal location for business is Albuquerque, New Mexico. The facility will be located in Lea County near Eunice, New Mexico. No other companies will be present or operating on the NEF site other than services specifically contracted by LES.

Foreign Ownership, Control and Influence (FOCI) of LES is addressed in the NEF Standard Practice Procedures for the Protection of Classified Matter, Appendix 1 – FOCI Package. The NRC in their letter dated, March 24, 2003, has stated "...that while the mere presence of foreign ownership would not preclude grant of the application, any foreign relationship must be examined to determine whether it is inimical to the common defense and security [of the United States]". (NRC, 2003) The FOCI Package mentioned above provides sufficient information for this examination to be conducted.

1.2.1.3 Address of the Enrichment Plant and Legal Site Description

The NEF is physically located approximately 8 km (5 mi) east of Eunice, New Mexico adjacent to New Mexico Highway 234 in Lea County. The legal description is as follows:

A PARCEL OF LAND WITHIN SECTION 32, TOWNSHIP 21 SOUTH, RANGE 38 EAST, NEW MEXICO PRINCIPAL MERIDIAN, LEA COUNTY, NEW MEXICO,

BEGINNING at the one-quarter corner between Sections 31 and 32, (a found GLO brass cap on a 2-in iron pipe);

THENCE N00°38'22"W along the section line between Sections 31 and 32 a distance of 2638.37 feet to the corner of Sections 29, 32, 31 and 30, (a found GLO brass cap on a 2-in iron pipe);

THENCE N89°18'08"E along the section line between Sections 29 and 32 a distance of 2640.69 feet to a set 5/8-in rebar with a 2-in aluminum cap marked "MUTH PLS 13239";

THENCE N89°18'08"E along the section line between Sections 29 and 32 a distance of 2640.69 feet to the corner of Sections 28, 33, 32 and 29, (a found GLO brass cap on a 2-in iron pipe);

THENCE S00°39'20"E along the section line between Sections 32 and 33 a distance of 2640.49 feet to the one-quarter corner between Sections 32 and 33, (a found GLO brass cap on a 1-in iron pipe);

THENCE S00°41'56"E along the section line between Sections 32 and 33 a distance of 2324.52 feet to a found railroad iron marking the right-of-way for New Mexico State Highway No. 234; from whence the corner of Sections 33 and 32 of Township 21 South, Range 38 East, and Sections 4 and 5 of Township 22 South, Range 38 East (a found 1/2-in rebar) bears S00°41'56"E a distance of 340.08 ft;

THENCE N80°10'49"W along the observed northerly right-of-way line of New Mexico State Highway No. 234 a distance of 5377.12 ft to a point of intersection with the section line between Sections 31 and 32 (set 5/8-in rebar with a 2-in aluminum cap marked "MUTH PLS 13239"); from whence the corner of Sections 31 and 32 of Township 21 South, Range 38 East, and Sections 6 and 5 of Township 22 South, Range 38 East (a found GLO brass cap on a 2-in iron pipe) bears S00°35'16"E a distance of 1321.66 ft;

THENCE N00°35'16"W along the section line between Sections 31 and 32 a distance of 1345.14 to the POINT OF BEGINNING

Said Parcel CONTAINS 542.80 ACRES more or less

1.2.2 Financial Information

LES estimates the total cost of the NEF to be approximately $1.2 billion (in 2002 dollars), excluding escalation, contingency, interest, tails disposition, decommissioning, and any replacement equipment required during the life of the facility.

There are financial qualifications to be met before a license can be issued. LES acknowledges the use of the following Commission-approved criteria as described in Policy Issues Associated with the Licensing of a Uranium Facility; Issue 3, Financial Qualifications (LES, 2002) in determining if the project is financially feasible:

1. Construction of the facility shall not commence before funding is fully committed. Of this full funding (equity and debt), the applicant must have in place before constructing the associated capacity: (a) a minimum of equity contributions of 30% of project costs from the parents and affiliates of the partners; and (b) firm commitments ensuring funds for the remaining project costs.
2. LES shall not proceed with the project unless it has in place long-term enrichment contracts (i.e., five years) with prices sufficient to cover both construction and operation costs, including a return on investment, for the entire term of the contracts.

LES shall in accordance with 10 CFR 140.13b, (CFR, 2003l), prior to and throughout operation, have and maintain nuclear liability insurance in the amount of up to $300 million to cover liability claims arising out of any occurrence within the United States, causing, within or outside the United States, bodily injury, sickness, disease, or death, or loss of or damage to property, or loss of use of property, arising out of or resulting from the radioactive, toxic, explosive, or other hazardous properties of chemical compounds containing source or special nuclear material.

The amounts of nuclear energy liability insurance required may be furnished and maintained in the form of:

1. An effective facility form (non-indemnified facility) policy of nuclear energy liability insurance from American Nuclear Insurers and/or Mutual Atomic Energy Liability underwriters; or
2. Such other type of nuclear energy liability insurance as the Commission may approve; or
3. A combination of the foregoing.

If the form of liability insurance will be other than an effective facility form (non-indemnified facility) policy of nuclear energy liability insurance from American Nuclear Insurers and/or Mutual Atomic Energy Liability Underwriters, such form will be provided to the Nuclear Regulatory Commission by LES. The effective date of this insurance will be no later than the date that LES takes possession of licensed nuclear material.

Effective November 26, 2002, nuclear energy liability Facility Form policy number NF-0350 was issued to LES for the planned NEF with the limit of liability of $1,000,000. This standby limit will apply until the plant takes possession of source or special nuclear material, at which time it is anticipated that the liability insurance coverage limit will be increased to more closely approximate the $300 million limit. Until such time as LES takes possession of source or special nuclear material, the effects described in 10 CFR 140.13b involving source or special nuclear material are not possible. Therefore, the $1,000,000 standby liability policy, in addition to appropriate construction coverage, is considered to be sufficient for the construction phase. LES will provide proof of liability insurance of a type and in the amounts to cover liability claims required by 10 CFR 140.13b prior to taking possession of source or special nuclear material.

Information indicating how reasonable assurance will be provided that funds will be available to decommission the facility as required by 10 CFR 70.22(a)(9) (CFR, 2003b), 10 CFR 70.25 (CFR, 2003c), and 10 CFR 40.36 (CFR, 2003d) is described in detail in Chapter 10, Decommissioning.

1.2.3 Type, Quantity, and Form of Licensed Material

LES proposes to acquire, deliver, receive, possess, produce, use, transfer, and/or store special nuclear material (SNM) meeting the criteria of *special nuclear material of low strategic significance* as described in 10 CFR 70.4 (CFR, 2003e). Details of the SNM are provided in Table 1.2-1, Type, Quantity, and Form of Licensed Material. It is expected that other source materials and by-product materials will also be used for instrument calibration purposes. These materials will be identified during the design phase and the SAR will be revised, accordingly.

1.2.4 Requested Licenses and Authorized Uses

LES is engaged in the production and selling of uranium enrichment services to electric utilities for the purpose of manufacturing fuel to be used to produce electricity in commercial nuclear power plants.

This application is for the necessary licenses issued under 10 CFR 70 (CFR, 2003f), 10 CFR 30 (CFR, 2003g) and 10 CFR 40 (CFR, 2003h) to construct, own, use and operate the facilities described herein as an integral part of the uranium enrichment facility. This includes licenses for source, special nuclear material and byproduct material. The period of time for which the license is requested is 30 years.

See Section 1.1, Facility and Process Description for a summary, non-technical narrative description of the enrichment activities utilized in NEF.

1.2.5 Special Exemptions or Special Authorizations

In accordance with 10 CFR 40.14 (CFR, 2005a), "Specific exemptions," and 10 CFR 70.17 (CFR, 2005b), "Specific exemptions," LES requests exemptions from certain provisions of 10 CFR 40.36 (CFR, 2005c), "Financial assurance and recordkeeping for decommissioning," paragraph (d), and 10 CFR 70.25 (CFR, 2005d), "Financial assurance and recordkeeping for decommissioning," paragraph (e). Specifically, 10 CFR 40.36(d) (CFR, 2005c) and 10 CFR 70.25(e) (CFR, 2005d) both state in part that "...the decommissioning funding plan must also contain a certification by the licensee that financial assurance for decommissioning has been provided in the amount of the cost estimate for decommissioning...." As stated in Section 10.2.1, "Decommissioning Funding Mechanism," of the SAR since LES intends to sequentially install and operate modules of the enrichment equipment over time, providing financial assurance for decommissioning during the operating life of the NEF at a rate that is in proportion to the decommissioning liability for these facilities as they are phased in satisfies the requirements of this regulation without imposing the financial burden of maintaining the entire financial coverage for facilities and material that are not yet in existence. The same basis applies to decommissioning funding assurance for depleted uranium byproduct. As also stated in Section 10.2.1 of the SAR, LES proposes to provide financial assurance for the disposition of depleted uranium byproduct at a rate in proportion to the amount of accumulated depleted uranium byproduct onsite up to the maximum amount of the depleted uranium byproduct produced by the NEF.

The justification for this proposal to provide decommissioning funding assurance on a forward-looking incremental basis is LES's commitment to update the decommissioning cost estimates and to provide to the NRC a revised funding instrument for facility decommissioning at a minimum prior to the operation of each facility module. With respect to the depleted uranium byproduct, LES commits to updating the decommissioning cost estimates on an annual forward-looking incremental basis and to providing the NRC revised funding instruments that reflect these projections of depleted uranium byproduct production. The long-term nature of enrichment contracts allows LES to accurately predict the production of depleted uranium byproduct. If any adjustments to the funding assurance were determined to be needed during the annual period due to production variations, they would be made promptly and a revised funding instrument would be provided to the NRC.

LES requests that exemptions from the provisions of 10 CFR 40.36(d) (CFR, 2005c) and 10 CFR 70.25(e) (CFR, 2005d) described above be granted. In support of this request, LES provides the following information relative to the criteria in 10 CFR 40.14 (CFR, 2005a) and 10 CFR 70.17 (CFR, 2005b).

Granting the exemption is authorized by law

There is no statutory prohibition to providing decommissioning funding assurance on an incremental basis. In fact, the NRC has previously accepted an incremental approach to decommissioning funding assurance for the United States Enrichment Corporation's operation of its gaseous diffusion plants.

Granting the exemptions will not endanger life or property or the common defense and security

Allowing the decommissioning funding assurance for the NEF to be provided on a forward-looking incremental basis continues to ensure that adequate funds are available at any point in time after licensed material is introduced onto the NEF site to decommission the facility and disposition any depleted uranium byproduct possessed by LES. Accordingly, life, property, or the common defense and security will not be endangered by the NEF once it is permanently shutdown.

Granting the exemptions is otherwise in the public interest

Providing an alternative, diverse, and secure domestic source of enrichment services in support of the nuclear power industry that supplies 20% of the nation's electricity is clearly in the public benefit. Providing decommissioning funding assurance on an incremental basis will ensure that adequate financial assurance is available when required. Imposing the requirement to provide decommissioning funding assurance for the entire facility and all depleted uranium byproduct that would be produced over the NEF licensed operating period results in a significant unnecessary financial hardship. Accordingly, the granting of these exemptions is in the public interest.

Since the granting of this exemption does not satisfy any of the criteria for categorical exclusion delineated in 10 CFR 51.22 (CFR, 2005e), "Criteria for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," nor the criteria requiring an environmental impact statement in 10 CFR 51.20 (CFR, 2005f), "Criteria for and identification of licensing and regulatory actions requiring environmental impact statements," an environmental assessment is required in accordance with 10 CFR 51.21 (CFR, 2005g), "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." Accordingly, LES proposes that the NRC make a finding of no significant impact based on the following information addressing the provisions of 10 CFR 50.30 (CFR, 2005h), "Environmental assessment."

Need for the proposed action

Granting of the requested exemption will allow LES to satisfy the applicable decommissioning funding assurance requirements for the NEF without imposing an unnecessary financial burden on LES.

Alternatives as required by Section 102(2)(E) of the National Environmental Policy Act (NEPA)

The only alternative to granting the requested exemption is to not grant it. The significant financial burden that would be imposed on LES by not granting the requested exemption is unnecessary.

The environmental impacts of the proposed action and alternatives as appropriate

Granting the requested exemption will not result in environmental impacts in addition to those delineated in the ER for the NEF since adequate funds will continue to be available to decommission the NEF and disposition any depleted uranium byproduct possessed by LES at any point in time after licensed material is introduced onto the NEF site. The environmental impact of not granting the requested exemption could potentially be the loss of an alternate, diverse, and secure domestic source of enrichment services for the nuclear power industry that supplies 20% of the nation's electricity.

A list of agencies and persons consulted and identification of sources used

The NRC Project Manager for the NEF was contacted. The NEF license application was used as a source.

Based on the above information, LES proposes that, if this exemption request is granted, the NRC reach a finding of no significant impact in accordance with 10 CFR 51.32 (CFR, 2005i), "Finding of no significant impact."

1.2.6 Security of Classified Information

Access to restricted data or national security information shall be controlled in accordance with 10 CFR 10 (CFR, 2003i), 25 (CFR, 2003j), and 95 (CFR, 2003k). This application does contain classified information that has been submitted under separate correspondence.

1.3 SITE DESCRIPTION

The NEF is located in southeastern New Mexico in Lea County near the border of Andrews County, Texas. The site consists of land north of New Mexico Highway 234 within Section 32 of Township 21 S, Range 38 E. The nearest communities are Eunice, about 8 km (5 mi) due west and Hobbs about 32 km (20 mi) north of the site. The area surrounding the site consists of vacant land and industrial properties. A railroad spur borders the site to the north. Further north is a sand/aggregate quarry operated by the Wallach Concrete Company. The quarry owner leases land space to a "produced water" reclamation company, Sundance Services, which maintains three small "produced water" lagoons. There is also a man-made pond stocked with fish on the quarry property.

A vacant parcel of land, Section 33, is immediately to the east. Section 33 borders the New Mexico/Texas state line that is 0.8 km (0.5 mi) east of the site. Several disconnected power poles are situated in front of Section 33, parallel to New Mexico Highway 234. Land further east, in Texas, is occupied by Waste Control Specialists (WCS), LLC. WCS possesses a radioactive materials license from Texas, an NRC Agreement state, and is licensed to treat and temporarily store low-level radioactive waste. Land east of WCS is occupied by the Letter B Ranch.

High powered utility lines run in a north-south direction near the property line of WCS, parallel to the New Mexico/Texas state line.

To the southeast, across New Mexico Highway 234, is the Lea County Landfill.

Land further north, south and west has mostly been developed by the oil and gas industry.

An underground CO_2 pipeline owned by Trinity Pipeline, LLC, running southeast-northwest, traverses the property. An underground natural gas pipeline owned by the Sid Richardson Energy Services Company is located along the south property line, paralleling New Mexico Highway 234.

An active railroad line, operated by the Texas-New Mexico Railroad, runs parallel to New Mexico Highway 18 and just east of Eunice within 8 km (5 mi) of Section 32. There is also an active railroad spur that runs from the Texas-New Mexico Railroad line, along the north boundary of Section 32 and terminates at the WCS facility.

Figure 1.3-1, Five Mile Radius, Radial Sectors, shows the physical features surrounding the facility to an 8 km (5 mi) radius.

1.3.1 Site Geography

Site features are well suited for the location of a uranium enrichment facility as evidenced by the favorable conditions of hydrology, geology, seismology and meteorology as well as good transportation routes for transporting feed and product by truck.

1.3.1.1 Site Location Specifics

The proposed 220 ha (543 acre) site is located within Section 32 of Township 21 S in southeastern New Mexico in Lea County approximately 0.8 km (0.5 mi) west of the Texas state border, 51 km (32 mi) west-north-west of Andrews, Texas and 523 km (325 mi) southeast of Albuquerque, New Mexico. This location is 8 km (5 mi) due east of Eunice and 32 km (20 mi) south of Hobbs. The geographic location of the facility is shown on Figures 1.1-1, State Map, and 1.1-2, County Map.

The approximate center of the NEF is at latitude 32 degrees, 26 minutes, 1.74 seconds North and longitude 103 degrees, 4 minutes, 43.47 seconds West. Section 32 is currently owned by the State of New Mexico and is being acquired by LES through a state land swap arrangement. Until the land swap is completed, LES has been granted a 35 year easement by the State of New Mexico for site access and control.

Figure 1.1-4, Facility Layout (Site Plan) with Site Boundary and Controlled Access Area Boundary, shows the site property boundary, including the Controlled Access Area and the general layout of the buildings.

1.3.1.2 Features of Potential Impact to Accident Analysis

The NEF site is located in the Pecos Plains Section of the Great Plains Province. Site topography is relatively level, with an overall gradual rise in elevation from the southwest to the northeast. An area comprised of small sand hills exists along the west property line. There are no mountain ranges in the immediate vicinity. Earthquakes in the region are isolated or occur in small clusters of low to moderate size events toward the Rio Grande Valley of New Mexico and southeast of the NEF site in Texas.

An underground natural gas pipeline owned by the Sid Richardson Energy Services Company is located along the south property line, paralleling New Mexico Highway 234.

An underground CO_2 pipeline owned by Trinity Pipeline, LLC, running southeast-northwest, currently traverses the property. This pipeline will be relocated to the NEF site property boundary.

New Mexico Highway 234 runs parallel to the southern property line. New Mexico Highway 234 intersects New Mexico Highway 18 about 4 km (2.5 mi) to the west.

An active railroad line operated by the Texas-New Mexico Railroad runs parallel to Highway 18 and just east of Eunice within 8 km (5 mi) of Section 32.

1.3.2 Demographics

This section provides the census results for the facility site area, and includes specific information about populations, public facilities (schools, hospitals, parks, etc.) and land and water use near the site.

1.3.2.1 Latest Census Results

The combined population of the two counties in the NEF vicinity, based on the 2000 U.S. Census is 68,515, which represents a 2.3% decrease from the 1990 population of 70,130. This decrease is counter to the trends for the states of New Mexico and Texas which had population increases of 20.1% and 22.8%, respectively during the same decade. Over that 10 year period, Lea County, New Mexico, where the site is located, had a growth decrease of 0.5%. The growth decrease in Andrews County, Texas was 9.3%. Lea County experienced a sharp but short population increase in the mid-1980's due to an influx of petroleum industry jobs. That influx caused its population to increase to over 65,000 during that period.

Based on projections made using historic data, the population of Lea County, New Mexico and Andrews County, Texas is likely to grow more slowly than their respective states over the next 30 years (the anticipated license period of the NEF).

Based on U. S. census data the minority populations of the Lea County New Mexico and Andrews County Texas as of 2000 were 32.9% and 22.9%, respectively. These percentages are consistent with their respective state averages of 34.7% and 26.4%.

The low income population of Lea County, New Mexico and Andrews County, Texas are 21.1% and 16.4% respectively. These percentages are consistent with their respective state averages of 18.4% and 15.4%. Within the site area the percentage of population below the poverty level is significantly lower in both states.

1.3.2.2 Description, Distance, And Direction To Nearby Population Areas

The NEF site is in Lea County, New Mexico near the border of Andrews County, Texas. The nearest community is Eunice, approximately 8 km (5 mi) east of the site. Other population centers are at distances from the site as follows:

- Hobbs, Lea County, New Mexico: 32 km (20 mi north)
- Jal, Lea County, New Mexico: 37 km (23 mi south)
- Lovington, Lea County New Mexico: 64 km (39 mi north-northwest)
- Andrews, Andrews County Texas: 51 km (32 mi east)
- Seminole, Gaines County Texas: 51 km (32 mi east-northeast)
- Denver City, Gaines County, Texas: 65 km (40 mi) north-northeast

Aside from these communities, the population density around the site is extremely low. The nearest large population center (>100,000) is Midland-Odessa, Texas which is approximately 103 km (64 mi) to the southeast.

1.3.2.3 Proximity to Public Facilities – Schools, Hospitals, Parks

The Eunice First Assembly of God Church is located about 9 km (5.4 mi) from the site.

There are two hospitals in the vicinity of the site. The Lea Regional Medical Center is located in Hobbs, New Mexico about 32 km (20 mi) north of the NEF site. This 250-bed hospital can handle acute and stable chronic care patients. In Lovington, New Mexico, 64 km (39 mi) north-northwest of the site, Covenant Medical Systems manages Nor-Lea Hospital, a full-service, 27-bed facility.

Eunice Senior Center is located about 9 km (5.4 mi) from the site.

There are four educational facilities within about 8 km (5 mi) of the NEF site, all in Eunice, New Mexico. These include an elementary school, a middle school, a high school, and a private K-12 school.

Eunice Fire and Rescue and the Eunice Police Department are located approximately 8 km (5 mi) from the site.

The Eunice Golf Course is located approximately 14.7 km (9.4 mi) from the site.

1.3.2.4 Nearby Industrial Facilities (Includes Nuclear Facilities)

Nuclear Facilities

There are no nuclear production facilities located within 32 km (20 mi) of the site, therefore neither environmental nor emergency preparedness interactions between facilities is required.

Non-Nuclear Facilities

The site is bordered to the north by railroad tracks beyond which is a quarry operated by Wallach Concrete Company. The quarry owner leases land space to Sundance Services, a reclamation company, that maintains three small "produced water" lagoons.

Lea County operates a landfill on the south side of Section 33 across New Mexico State Highway 234, approximately 1 km (0.6 mi) from the center of the site.

A vacant parcel of land is immediately east of the site. Land further east, in Texas, is occupied by WCS. WCS possesses a radioactive materials license from Texas, an NRC Agreement state, and is licensed to treat and temporarily store low-level radioactive waste.

Dynegy's Midstream Services Plant is located 6 km (4 mi) from the site. This facility is engaged in the gathering and processing of natural gas for the subsequent fractionation, storage, and transportation of natural gas liquids.

An underground CO_2 pipeline, running southeast-northwest, currently traverses the property.

An underground natural gas pipeline is located along the south property line, paralleling New Mexico Highway 234.

Eunice maintains water supply tanks approximately 8 km (5 mi) north and 8 km (5 mi) south of the site.

Land further north, south and west of the site has mostly been developed by the oil and gas industry.

The Eunice Airport is situated about 8 km (5 mi) west of the town center. The nearest commercial carrier airport is Lea County Regional Airport in Hobbs, New Mexico about 40 km (25 mi) north-northwest of the site. A major commercial airport in Midland-Odessa, Texas is approximately 103 km (64 mi) to the southeast.

1.3.2.5 Land Use Within Eight Kilometers (Five Mile) Radius, Uses Of Nearby Bodies Of Water

The site and vicinity are within the southern part of the Llano Estacado or Staked Plains, which is a remnant of the Southern High Plains. The site area overlies prolific oil and gas geologic formations of the Pennsylvanian and Permian age.

Onsite soils consist of fine sand, loamy fine sand and loose sands surrounding large barren sand dunes and are common to areas used for rangeland and wildlife habitat.

Surrounding property consists of vacant land and industrial developments. Gas and oil field operations are widespread in the area, but significant petroleum potential is absent within 5 to 8 km (3 to 5 mi) of the site.

More than 98% of the area within an 8 km (5 mi) radius of the NEF is an extensive area of open land on which livestock wander and graze. Built-up land (1.2%) and barren land (0.3%) constitute the other two land use classifications in the site vicinity.

Baker Spring, an intermittent surface water feature, is situated a little over 1.6 km (1 mi) northeast of the NEF site.

The facility will make no use of either surface water or groundwater supply from the site. A site Septic System and a Site Stormwater Detention Basin will discharge to the ground with a Groundwater Discharge Permit/Plan from the New Mexico Water Quality Bureau. No significant adverse changes are expected in site hydrology as a result of construction or operation of the NEF. Section 4, Environmental Impacts, of the Environmental Report addresses potential for impacts on site hydrology as a result of activities on the site.

1.3.3 Meteorology

In this section, data characterizing the meteorology (e.g., winds, precipitation, and severe weather) for the site are presented.

1.3.3.1 Primary Wind Directions And Average Wind Speeds

The meteorological conditions at the NEF have been evaluated and summarized in order to characterize the site climatology and to provide a basis for predicting the dispersion of gaseous effluents.

Meteorological data from the National Weather Service (NWS) site at Midland-Odessa, Texas, indicate an annual mean wind speed of 4.9 m/s (11.0 mi/hr). The prevailing wind direction is wind from the south. The maximum five-second wind speed is 31.3 m/sec (70 mph) from 200 degrees with respect to true north.

By comparison, the data from Roswell, New Mexico indicate the annual mean wind speed is 3.7 m/s (8.2 mi/hr) and the prevailing wind direction is wind from the south-southeast. The maximum five-second wind speed is 27.7 m/sec (62 mph) from 270 degrees with respect to true north.

These and additional data are discussed and further analyzed in the Environment Report.

1.3.3.2 Annual Precipitation – Amounts and Forms

The NEF site is located in the Southeast Plains of New Mexico near the Texas border. The climate is typical of a semi-arid region, with generally mild temperatures, low precipitation and humidity, and a high evaporation rate. Vegetation consists mainly of native grasses and some mesquite trees. During the winter, the weather is often dominated by a high-pressure system located in the central part of the western United States and a low-pressure system located in north-central Mexico. During the summer, the region is affected by a low-pressure system normally located over Arizona.

The normal annual total rainfall as measured in Hobbs, New Mexico is 46.1 cm (18.15 in). Precipitation amounts range from an average of 1.22 cm (0.48 in) in March to 7.95 cm (3.13 in) in September. Record maximum and minimum monthly totals are 35.13 cm (13.83 in) and zero respectively. (WRCC, 2003)

The normal annual total rainfall in Midland-Odessa, Texas, is 37.6 cm (14.8 in). Precipitation amounts range from an average of 1.1 cm (0.42 in) in March to 5.9 cm (2.31 in) in September. Record maximum and minimum monthly totals are 24.6 cm (9.70 in) and zero, respectively. The highest 24-hour precipitation total was 15.2 cm (5.99 in) in July 1968 (NOAA, 2002a).

The normal annual rainfall total as measured in Roswell, New Mexico, is 33.9 cm (13.34 in). Record maximum and minimum monthly totals are 17.50 cm (6.88 in) and zero , respectively (NOAA, 2002b, 2002a). The highest 24-hour precipitation total was 12.47 cm (4.91 in) in July 1981 (NOAA, 2002b).

Snowfall in Midland-Odessa, Texas, averages 13.0 cm (5.1 in) per year. Maximum monthly snowfall/ice pellets of 24.9 cm (9.8 in) fell in December 1998. The maximum amount of snowfall/ice pellets to fall in 24 hours was 24.9 cm (9.8 in) in December 1998 (NOAA, 2002a).

Snowfall in Roswell, New Mexico averages 30.2 cm (11.9 in) per year. Maximum monthly snowfall/ice pellets of 53.3 cm (21.0 in) fell in December 1997. The maximum amount of snowfall/ice pellets to fall in 24 hours was 41.91 cm (16.5 in) in February 1988 (NOAA, 2002b).

Additional details on rainfall and snowfall are provided in the Environmental Report.

The design basis snow load was developed using the methodology prescribed in the NRC Site Analysis Branch Position for Winter Precipitation Loads (NRC, 1975). The prescribed load to be included in the combination of normal live loads is based on the weight of the 100 year snowfall or snowpack whichever is greater. The winter precipitation load to be included in the combination of extreme live loads is based on the sum of the weight of the 100 year snowpack and the weight of the 48 hour Probable Maximum Winter Precipitation (PMWP) for the month corresponding to the selected snowpack.

The 100 year mean recurrence ground snow load was calculated to be 58.5 kg/m^2 (12 lb/ft^2), and the applicable PMWP was calculated to be 96.6 kg/m^2 (19.8 psf). The addition of these two figures results in a design load of 155.1 kg/m^2 (32 lb/ft^2).

1.3.3.3 Severe Weather

Tornadoes

Tornadoes occur infrequently in the vicinity of the NEF. Only two tornadoes were reported in Lea County, New Mexico, (Grazulis, 1993) from 1880-1989. Across the state line, only one tornado was reported in Andrews County, Texas, (Grazulis, 1993) from 1880-1989.

Tornadoes are commonly classified by their intensities. The F-Scale classification of tornados is based on the appearance of the damage that the tornado causes. There are six classifications, F0 to F5, with an F0 tornado having winds of 61-116 km/hr (40-72 mi/hr) and an F5 tornado having winds of 420-520 km/hr (261-318 mi/hr) (AMS, 1996). The two tornadoes reported in Lea County were estimated to be F2 tornadoes (Grazulis, 1993).

The design parameters applicable to the design tornado with a period of recurrence of 100,000 years are as follows:

Design Wind Speed	302 km/hr	188 mi/hr
Radius of damaging winds	130 m	425 ft
Atmospheric pressure change (APC)	390 kg/m^2	80 lb/ft^2
Rate of APC	146 kg/m^2/s	30 lb/ft^2/s

Hurricanes

Hurricanes, or tropical cyclones, are low-pressure weather systems that develop over the tropical oceans. Hurricanes are fueled by the relatively warm tropical ocean water and lose their intensity quickly once they make landfall. Since the NEF is located about 805 km (500 mi) from the coast, it is most likely that any hurricane that tracked towards the site would have dissipated to the tropical depression stage, that is, wind speeds less than 63 km/hr (39 mi/hr), before it reached the NEF. Hurricanes are therefore not considered a threat to the NEF.

Thunderstorms and Lightning Strikes

Thunderstorms occur during every month but are most common in the spring and summer months. Thunderstorms occur an average of 36.4 days/year in Midland/Odessa (based on a 54-year period of record (NOAA, 2002a). The seasonal averages are: 11 days in spring (March through May); 17.4 days in summer (June through August); 6.7 days in fall (September through November); and 1.3 days in winter (December through February).

The current methodology for estimating lightning strike frequencies includes consideration of the attractive area of structures (Marshall, 1973). This method consists of determining the number of lightning flashes to earth per year per square kilometer and then defining an area over which the structure can be expected to attract a lightning strike.

Using this methodology, the attractive area of the facility structures has been conservatively determined to be 0.071 km^2. Using 4 flashes to earth per year per square kilometer (2.1 flashes to earth per year per square mile) (NWS, 2003b) it can be estimated that the NEF will experience approximately 1.36 flashes to earth per year.

Sandstorms

Blowing sand or dust may occur occasionally in the area due to the combination of strong winds, sparse vegetation, and the semi-arid climate. High winds associated with thunderstorms are frequently a source of localized blowing dust. Dust storms that cover an extensive region are rare, and those that reduce visibility to less than 1.61 km (1 mile) occur only with the strongest pressure gradients such as those associated with intense extratropical cyclones which occasionally form in the area during winter and early spring (DOE, 2003).

1.3.4 Hydrology

The hydrology information presented for the NEF was based on a subsurface investigation initiated at the NEF site in September 2003. Extensive subsurface investigations for a nearby facility, WCS, located to the east of the NEF site, have also provided hydrogeologic data that was used in planning the NEF surface investigation. Other literature searches were also conducted to obtain reference material.

The NEF site itself contains no surface water bodies or surface drainage features. Essentially all the precipitation that occurs at the site is subject to infiltration and/or evapotranspiration. Groundwater was encountered at depths of 65 to 68 m (214 to 222 ft). Significant quantities of groundwater are only found at depths over 340 m (1,115 ft) where cover for that aquifer is provided by 323 to 333 m (1,060 to 1,092 ft) or more of clay.

1.3.4.1 Characteristics Of Nearby Rivers, Streams, And Other Bodies Of Water

The climate in southeast New Mexico is semi-arid. Precipitation averages only 33 to 38 cm (13 to 15 in) a year. Evaporation and transpiration rates are high. This results in minimal, if any surface water occurrence or groundwater recharge.

The NEF site contains no surface drainage features, such as arroyos or buffalo wallows. The site topography is relatively flat. Some localized depressions exist, due to eolian processes, but the size of these features is too small to be of significance with respect to surface water collection.

1.3.4.2 Depth To The Groundwater Table

The site subsurface investigation performed during September 2003 had two main objectives: 1) to delineate the depth to the top of the Chinle Formation red bed clay that exists beneath the NEF site to assess the potential for saturated conditions above the red beds, and 2) to complete three monitoring wells in the siltstone layer beneath the red beds to monitor water level and water quality within this thin horizon of perched intermittent saturation. This work is in progress as discussed below.

The presence of the thick Chinle clay beneath the site essentially isolates the deep and shallow hydrologic systems. Groundwater occurring within the red bed clay occurs at three distinct and distant elevations. Approximately 65 to 68 m (214 to 222 ft) beneath the land surface, within the red bed unit, is a siltstone or silty sandstone unit with some saturation. It is a low permeability formation that does not yield groundwater very readily. This unit is under investigation as the first occurrence of groundwater beneath the NEF site.

The next water bearing unit below the saturated siltstone horizon is a saturated 30.5-meter (100-foot) thick sandstone horizon approximately 183 m (600 ft) below land surface, which overlies the Santa Rosa formation. The Santa Rosa formation is the third water bearing unit and is located about 340 m (1,115 ft) below land surface. Between the siltstone and sandstone saturated horizons and the Santa Rosa formation lie a number of layers of sandstones, siltstones, and shales. Hydraulic connection between the siltstone and sandstone saturated horizons and the Santa Rosa formation is non-existent.

No withdrawals or injection of groundwater will be made as a result of operation of the NEF facility. Thus, there will be no affect on any inter-aquifer water flow.

1.3.4.3 Groundwater Hydrology

The climate in southeast New Mexico is semi-arid, and evapotranspiration processes are significant enough to short-circuit any potential groundwater recharge. There is some evidence for shallow (near-surface) groundwater occurrence in areas to the north at the Wallach Concrete plant. These conditions are intermittent and limited. The typical geologic cross section at that location consists of a layer of caliche at the surface, referred to as the "caprock." In some areas the caprock is missing and the sand and gravel are exposed at the surface. The caprock is generally fractured and, following precipitation events may allow infiltration that quickly bypasses any roots from surface vegetation. In addition, there are areas where the sand and gravel outcrop may allow rapid infiltration of precipitation. These conditions have led to instances of minor amounts of perched groundwater at the base of the sand and gravel unit, atop the red beds of the Chinle Formation.

Conditions at the NEF site are different than at the Wallach Concrete site. The caprock is not present at the NEF site. Therefore, rapid infiltration through fractured caliche does not contribute to localized recharge at the NEF site.

Another instance of possible saturation above the Chinle clay may be seen at Baker Spring, just to the northeast of the NEF site where the caprock ends. The surface water is intermittent, and water typically flows from Baker Spring only after precipitation events. Some water may seep from the sand and gravel unit beneath the caprock, but deep infiltration of water is impeded by the low permeability of the Chinle clay in the area. This condition does not exist at the NEF site due to the absence of the caprock and the low permeability surface soils.

A third instance of localized shallow groundwater occurrence exists to the east of the NEF site where several windmills on the WCS property were formerly used to supply water for live stock tanks. These windmills tapped small saturated lenses above the Chinle Formation red beds, but the amount of groundwater in these zones was limited.

1.3.4.4 Characteristics Of The Uppermost Aquifer

The first occurrence of a well-defined aquifer is approximately 340 m (1,115 ft) below land surface, within the Santa Rosa formation. No impacts are expected to the aquifer from the NEF because of the depth of the Santa Rosa formation, the thick Chinle clay overburden, and the fact that the NEF will not consume surface or groundwater or discharge to the surrounding area.

Treated liquid effluents are discharged to the onsite Treated Effluent Evaporative Basin, a double-lined evaporative basin with leak detection.

1.3.4.5 Design Basis Flood Events Used For Accident Analysis

The closest water conveyance is Monument Draw, a typically dry, intermittent stream located about 4 km (2.5 mi) west of the site. Since there are no bodies of water in the immediate vicinity of the site, flood is not a design basis event for the NEF. Additionally a diversion ditch is strategically located to deflect surface runoff from adjacent land away from the facility structures on the site.

The only potential flooding of the plant results from local intense rainfall. Flood protection against the local Probable Maximum Precipitation (PMP) is provided by establishing the facility floor level above the calculated depth of ponded water caused by the local PMP.

1.3.5 Geology

This section provides information about the characteristics of soil types and bedrock of the NEF site and its vicinity and design-basis earthquake magnitudes and return periods. The WCS site in Texas and the former proposed Atomic Vapor Laser Isotope Separation (AVLIS) site, located in Section 33, have both been thoroughly studied in recent years in preparation for construction of other facilities. A review of those documents and related materials provides a significant description of geological conditions pertinent to the NEF site. In addition, LES performed field confirmation, where necessary, in order to clarify any questions about regional or site-specific conditions.

The NEF site is located in New Mexico immediately west of the Texas border about 48 km (30 mi) from the extreme southeast corner of the state and about 96 km (80 mi) east of the Pecos River. The site is contained in the Eunice NE, Texas-New Mexico USGS topographic quadrangle (USGS, 1979). This location is near the boundary between the Pecos Plains Section to the west; and the Southern High Plains Section of the Great Plains province to the east. The boundary between the two sections is the Mescalero Escarpment, locally referred to as Mescalero Ridge.

NEF site elevations range between +1033 and +1045 m (+3390 and +3430 ft) (msl). The finished site grade is about +1041 m (+3415 ft) msl .

Surface exposures of geologic units at the site include surficial eolian deposits and Tertiary-aged alluvium. These overlie Triassic red-bed clay which overlies sedimentary rock. The principal underlying geologic structure is the Central Basin Platform which divides the Permian Basin into the Midland and Delaware sub-basins.

1.3.5.1 Characteristics Of Soil Types And Bedrock

The dominant subsurface structural feature of this region is the Permian Basin. This 250 million-year-old feature is the source of the Region's prolific oil and gas reserves.

The NEF site is located within the Central Permian Basin Platform area, where the top of the Permian deposits are approximately 434 to 480 m (1,425 to 1,575 ft) below ground surface. Overlying the Permian are the sedimentary rocks of the Triassic Age Dockum Group.

Soil development in the region is generally limited due to its semi-arid climate. The site has a minor thickness of soil (generally less than 0.4 m (1.4 ft)) developed from subaerial weathering. A small deposit of active dune sand is present at the southwest corner of the site. The U. S. Department of Agriculture soil survey for Lea County, New Mexico (USDA, 1974) categorizes site soils as hummocky loamy (silty) fine sand with moderately rapid permeability and slow runoff, well-drained non-calcareous loose sand, active dune sand and dune-associated sands.

Recent deposits are primarily dune sands derived from Permian and Triassic rocks of the Permian Basin. These Mescalero (dune) Sands cover over 80% of Lea County and are generally described as fine to medium-grained and reddish brown in color. The USDA Soil Survey of Lea County identifies the dune sands at the site as either the Brownsfield-Springer Association of reddish brown fine to loamy fine sands; or the Gomez series of brown to yellowish brown loamy fine sand (USDA, 1974).

1.3.5.2 Earthquake Magnitudes And Return Periods

The majority of earthquakes in the United States are located in the tectonically active western portion of the country. However, areas within New Mexico and the southwestern United States also experience earthquakes, although at a lower rate and at lower intensities. Earthquakes in the region around the NEF site include isolated and small clusters of low to moderate size events toward the Rio Grande Valley of New Mexico and in Texas, southeast of the NEF site.

The largest earthquake within 322 km (200 mi) of the NEF is the August 16, 1931 earthquake located near Valentine, Texas. This earthquake has an estimated magnitude of 6.0 to 6.4 and produced a maximum epicentral intensity of VIII on the Modified Mercalli Intensity (MMI) Scale. The intensity observed at the NEF site is IV on the MMI scale.

A site-specific probabilistic seismic hazard analysis was performed for the NEF site using the seismic source zone geometries and earthquake recurrence models. The modeling included attenuation models suited for the regional and local seismic wave transmission characteristics.

Total seismic ground motion hazard to a site results from summation of ground motion effects from all distant and local seismically active areas. The 250-year and 475-year return period peak horizontal ground accelerations are estimated at 0.024 g and 0.036 g, respectively. The 10,000 year return period peak horizontal ground acceleration is estimated at 0.15 g. This return period is equivalent to a mean annual probability of E-4. The associated peak vertical ground motion is also estimated at 0.15 g.

1.3.5.3 Other Geologic Hazards

There are no other known geologic hazards that would adversely impact the NEF site.

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TABLES

Table 1.1-1 Estimated Annual Gaseous Effluent
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Area	Quantity	Discharge m^3 (ft^3)
Gaseous Effluent Vent Systems	NA	2.6×10^8 @ Standard Temperature and Pressure (STP) (9.18×10^9)
HVAC Systems		
Radiological Areas	NA	1.5×10^9 (5.17×10^{10})
Non-Radiological Areas	N/A	1.0×10^9 (3.54×10^{10})
Total Gaseous HVAC Discharge	NA	2.47×10^9 (8.71×10^{10})
Constituents:		
Helium	440 m^3 @ (STP) (15,536 ft^3)	NA
Nitrogen	52 m^3 @ (STP) (1,836 ft^3)	NA
Ethanol	40 L (10.6 gal)	NA
Laboratory Compounds	Traces (HF) (NA)	NA
Argon	190 m^3 (6,709 ft^3)	NA
Hydrogen Fluoride	< 1.0 kg (< 2.2 lb)	NA
Uranium	< 10 g (< 0.0221 lb)	NA
Methylene Chloride	610 L (161 gal)	NA

Table 1.1-2 Estimated Annual Radiological and Mixed Wastes

Waste Type	Radiological Waste		Mixed Waste[1]	
	Total Mass kg (lb)	Uranium Content kg (lb)	Total Mass kg (lb)	Uranium Content kg (lb)
Activated Carbon	300 (662)	25 (55)	-	-
Activated Alumina	2160 (4763)	2.2 (4.9)	-	-
Fomblin Oil Recovery Sludge	20 (44)	5 (11)	-	-
Liquid Waste Treatment Sludge	400 (882)	57 (126)	-	-
Activated Sodium Fluoride[2]	-	-	-	-
Assorted Materials (paper, packing, clothing, wipes, etc.)	2100 (4,631)	30 (66)	-	-
Ventilation Filters	61,464 (135,506)	5.5 (12)		-
Non-Metallic Components	5000 (11,025)	Trace[3]	-	-
Miscellaneous Mixed Wastes (organic compounds)[4]			50 (110)	2 (4.4)
Combustible Waste	3,500 (7,718)	Trace[3]	-	-
Scrap Metal	12,000 (26,460)	Trace[3]	-	-

[1] A mixed waste is a low-level radioactive containing listed or characteristic of hazardous wastes as specified in 40 CFR 261, Subparts C and D.
[2] No sodium fluoride (NaF) wastes are produced on an annual basis. The contingency dump system NaF traps are not expected to saturate over the life of the plant.
[3] Trace is defined as not detectable above naturally occurring background concentrations.
[4] Representative organic compounds consist of acetone, toluene, ethanol, and petroleum ether.

Table 1.1-3 Estimated Annual Liquid Effluent
Page 1 of 1

Effluent	Typical Annual Quantities	Typical Uranic Content
Contaminated Liquid Process Effluents:	**m^3 (gal)**	**kg (lb)**
Laboratory Effluent/Floor Washings/Miscellaneous Condensates	23.14 (6,112)	16 (35)[1]
Degreaser Water	3.71 (980)	18.5 (41)[1]
Spent Citric Acid	2.72 (719)	22 (49)[1]
Laundry Effluent	405.8 (107,213)	0.2 (0.44)[2]
Hand Wash and Showers	2,100 (554,802)	None
Total Contaminated Effluent :	2,535 (669,884)	56.7 (125)[3]
Cooling Tower Blowdown:	19,123 (5,051,845)	None
Heating Boiler Blowdown:	138 (36,500)	None
Sanitary:	7,253 (1,916,250)	None
Stormwater Discharge:		
Gross Discharge[4]	174,100 (46 E+06)	None

[1]Uranic quantities are before treatment, values for degreaser water and spent citric acid include process tank sludge.
[2] Laundry uranic content is a conservative estimate.
[3] Uranic quantity is before treatment. After treatment approximately 1% or 0.57 kg (1.26 lb) of uranic material is expected to be discharged into the Treated Effluent Evaporative Basin.
[4]Maximum gross discharge is based on total annual rainfall on the site runoff areas contributing runoff to the Site Stormwater Detention Basin and the UBC Storage Pad Retention Basin neglecting evaporation and infiltration.

Table 1.1-4 Estimated Annual Non-Radiological Wastes
Page 1 of 1

Waste	Annual Quantity
Spent Blasting Sand*	125 kg (275 lbs)
Miscellaneous Combustible Waste*	9000 kg (19,800 lbs)
Cutting Machine Oils	45 L (11.9 gal)
Spent Degreasing Water (from ME&I workshop)	1 m^3 (264 gal)
Spent Demineralizer Water (from ME&I workshop)	200 L (53 gal)
Empty Spray Paint Cans*	20 ea
Empty Cutting Oil Cans	20 ea
Empty Propane Gas Cylinders*	5 ea
Acetone*	27 L (7.1 gal)
Toluene*	2 L (0.5 gal)
Degreaser Solvent SS25*	2.4 L (0.6 gal)
Petroleum Ether*	10 L (2.6 gal)
Diatomaceous Earth*	10 kg (22 lbs)
Miscellaneous Scrap metal	2,800 kg (6.147 lbs)
Motor Oils (For internal combustion. engines)	3,400 L (895 gal)
Oil Filters	250 ea
Air Filters (vehicles)	50 ea
Air Filters (building ventilation)	160,652 kg (354,200 lb)
Hydrocarbon Sludge*	10 kg (22 lbs)
Methylene Chloride*	1850 L (487 gal)

* Hazardous waste as defined in Title 40, Code of Federal Regulations, Part 261, Identification and listing of hazardous waste, 2003. (in part or whole)

Table 1.1-5 Annual Hazardous Construction Wastes

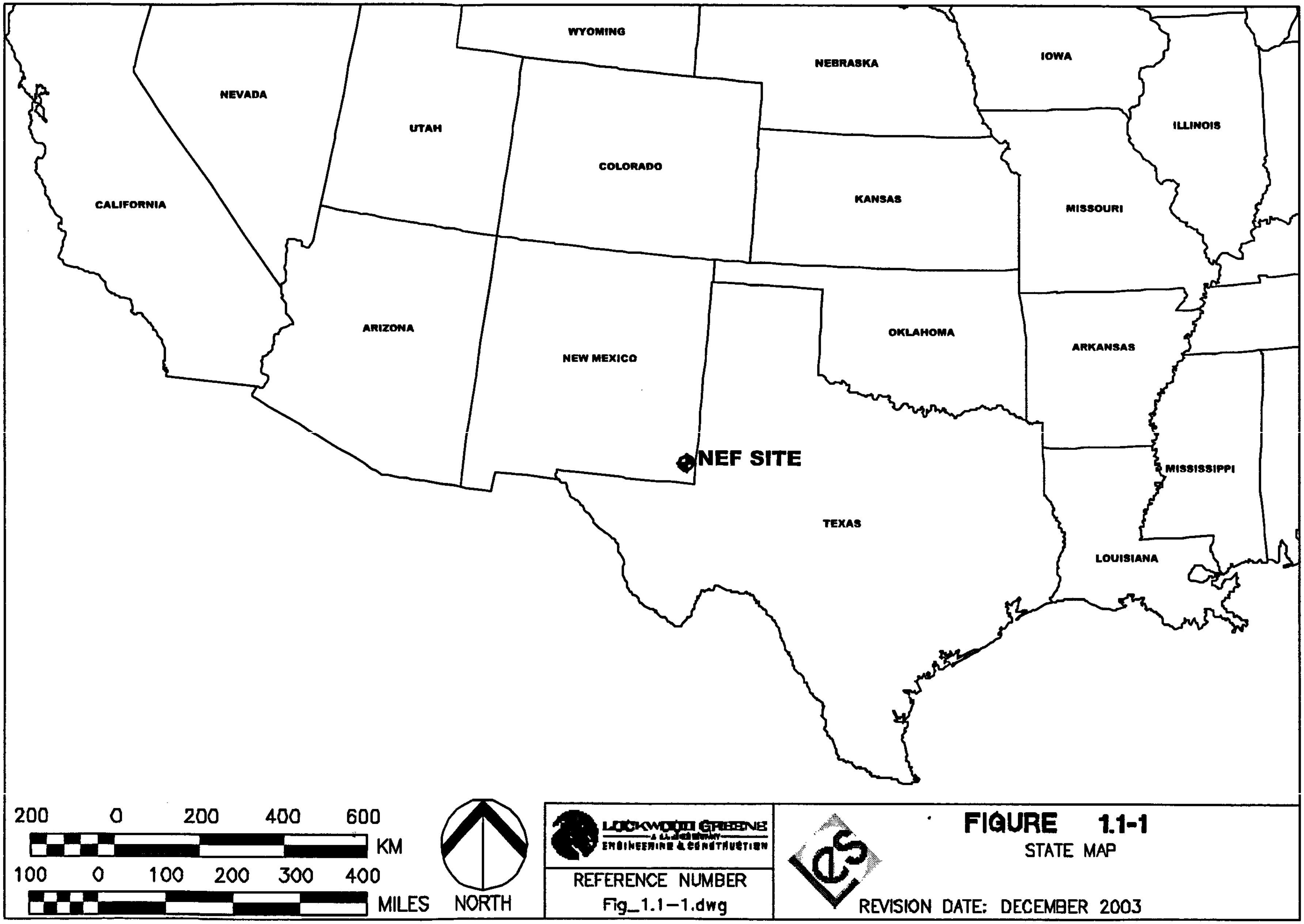
Waste Type	Annual Quantity
Paint, Solvents, Thinners, Organics	1,134 L (3,000 gal)
Petroleum Products – Oils, Lubricants	1,134 L (3,000 gal)
Sulfuric Acid (Batteries)	380 L (100 gal)
Adhesives, Resins, Sealers, Caulking	910 kg (2,000 lbs)
Lead (Batteries)	91 kg (200 lbs)
Pesticide	380 L (100 gal)

Table 1.2-1 Type, Quantity and Form of Licensed Material
Page 1 of 1

Source and/or Special Nuclear Material	Physical and Chemical Form	Maximum Amount to be Possessed at Any One Time
Uranium (natural and depleted) and daughter products	Physical: Solid, Liquid and Gas Chemical: UF_6, UF_4, UO_2F_2, oxides and other compounds	136,120,000 kg
Uranium enriched in isotope ^{235}U up to 5% by weight and uranium daughter products	Physical: Solid, Liquid, and Gas Chemical: UF_6, UF_4, UO_2F_2, oxides and other compounds	545,000 kg
^{99}Tc, transuranic isotopes and other contamination	Any	Amount that exists as contamination as a consequence of the historical feed of recycled uranium at other facilities(1)

(1) To minimize potential sources of contamination of UF_6, such as ^{99}Tc, LES will require UF_6 suppliers to provide Commercial Natural UF_6 in accordance with ASTM C 787-03, "Standard Specification for Uranium Hexafluoride for Enrichment." In addition, cylinder suppliers will be required to preclude use of cylinders that, in the past, have contained reprocessed UF_6, unless they have been decontaminated. Periodic audits of suppliers will be performed to provide assurance that these requirements are satisfied.

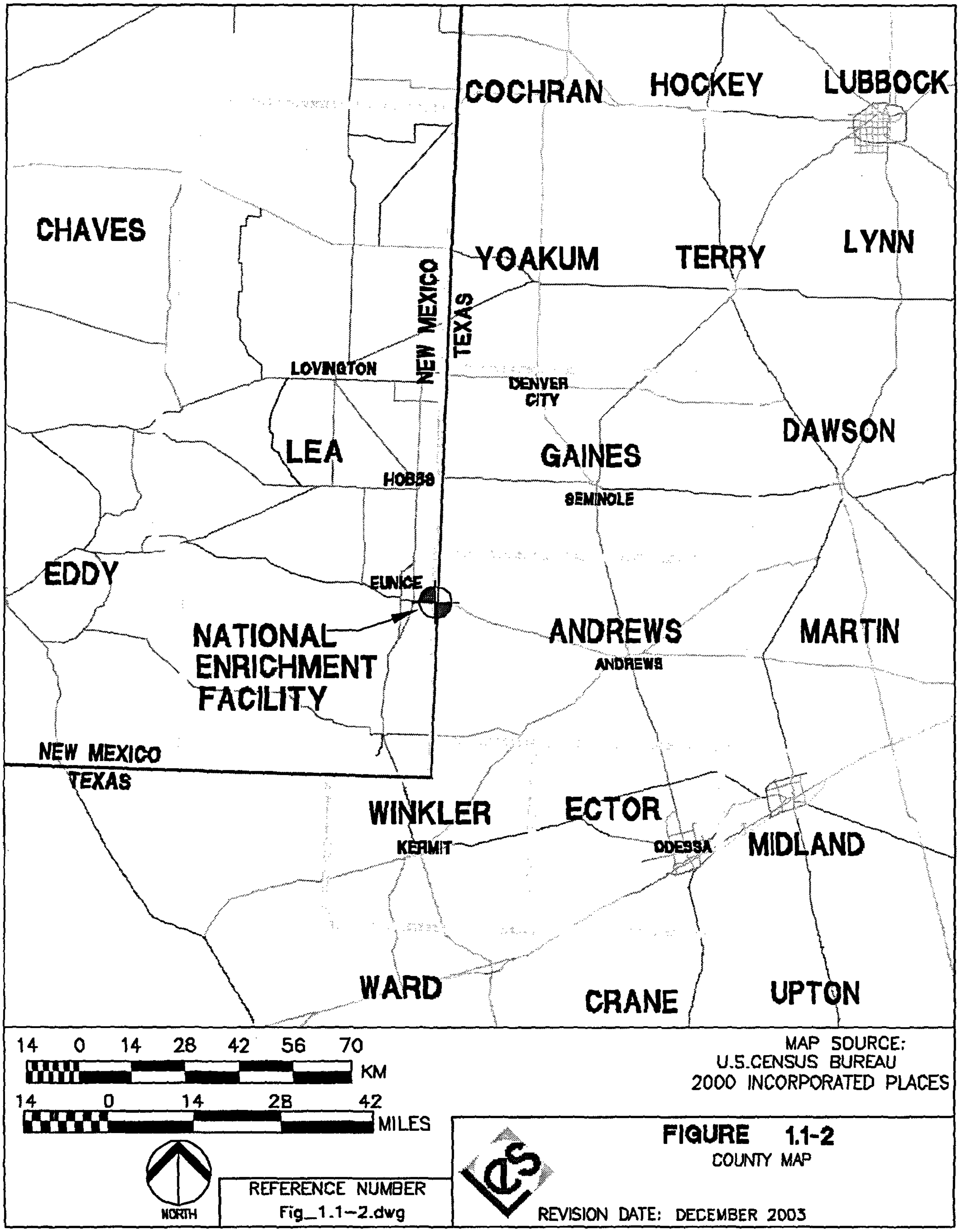
FIGURES



REFERENCE NUMBER
Fig_1.1-1.dwg

FIGURE 1.1-1
STATE MAP

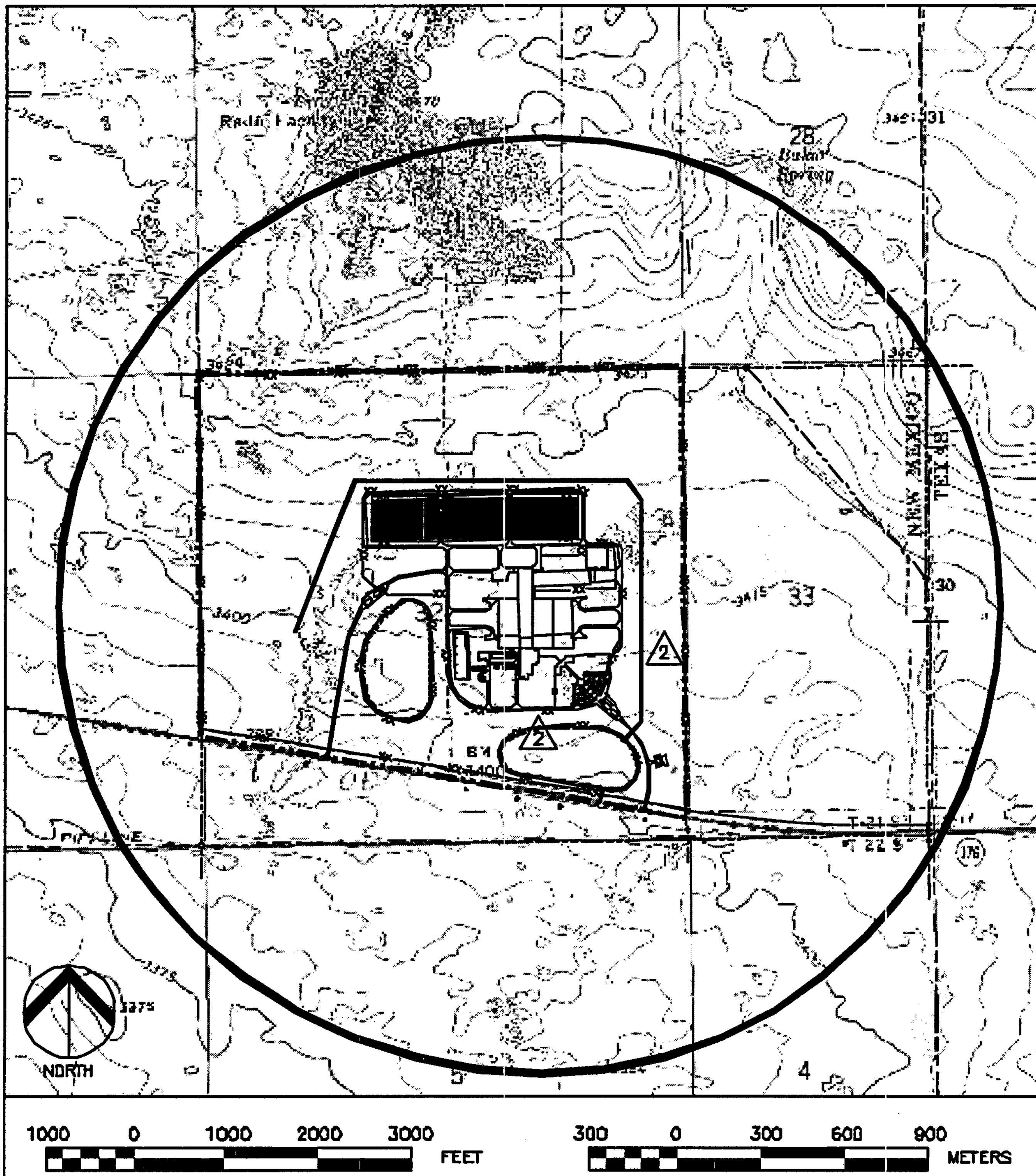
REVISION DATE: DECEMBER 2003



REFERENCE NUMBER
Fig_1.1-2.dwg

FIGURE 1.1-2
COUNTY MAP

REVISION DATE: DECEMBER 2003



MAP SOURCE:
USGS 7.5 MINUTE
EUNICE NE QUADRANGLE
TEX.-N. MEX. 1:24000

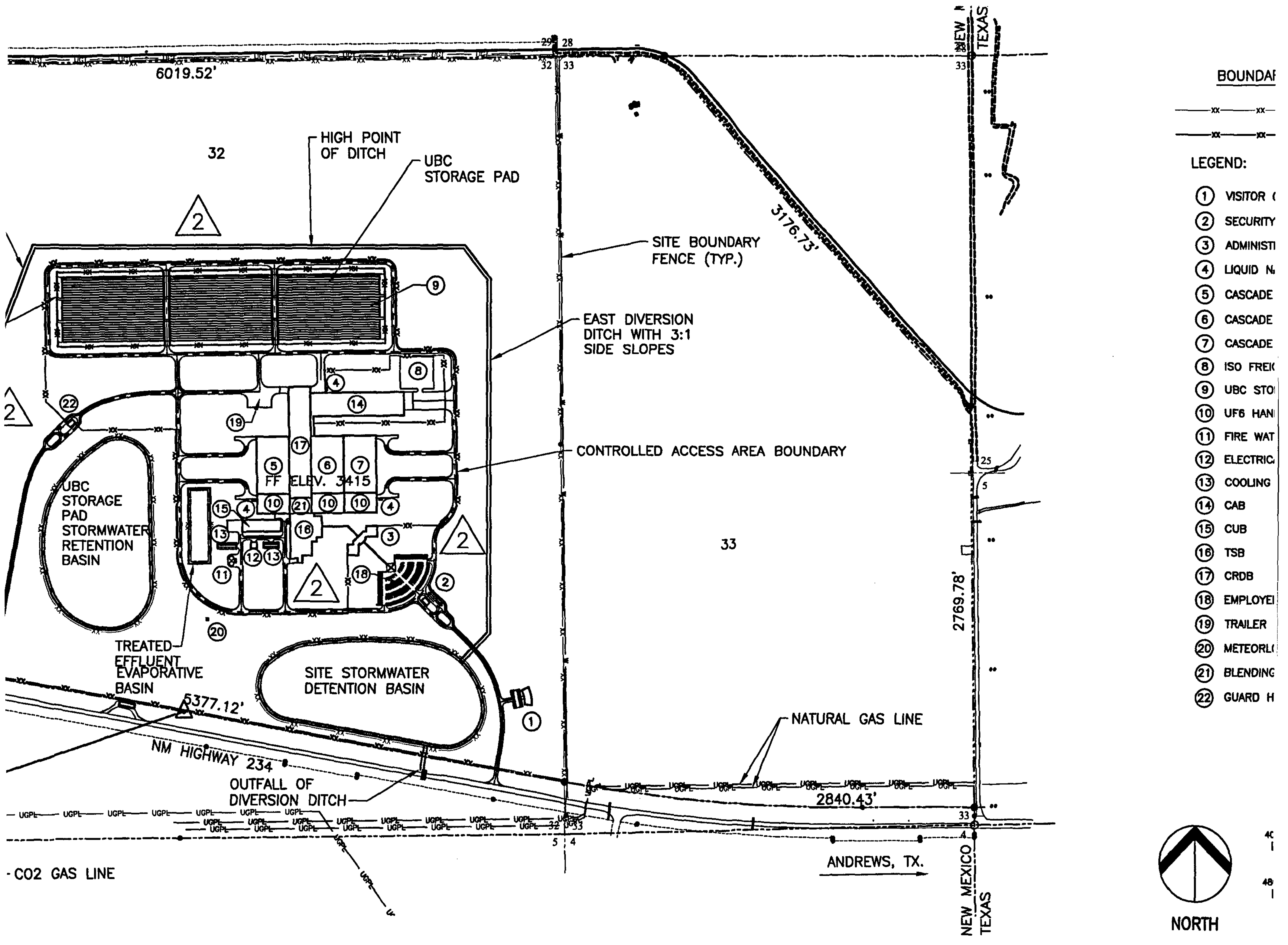
CONTOUR INTERVAL:
5 FEET

LOCKWOOD GREENE ENGINEERING & CONSTRUCTION

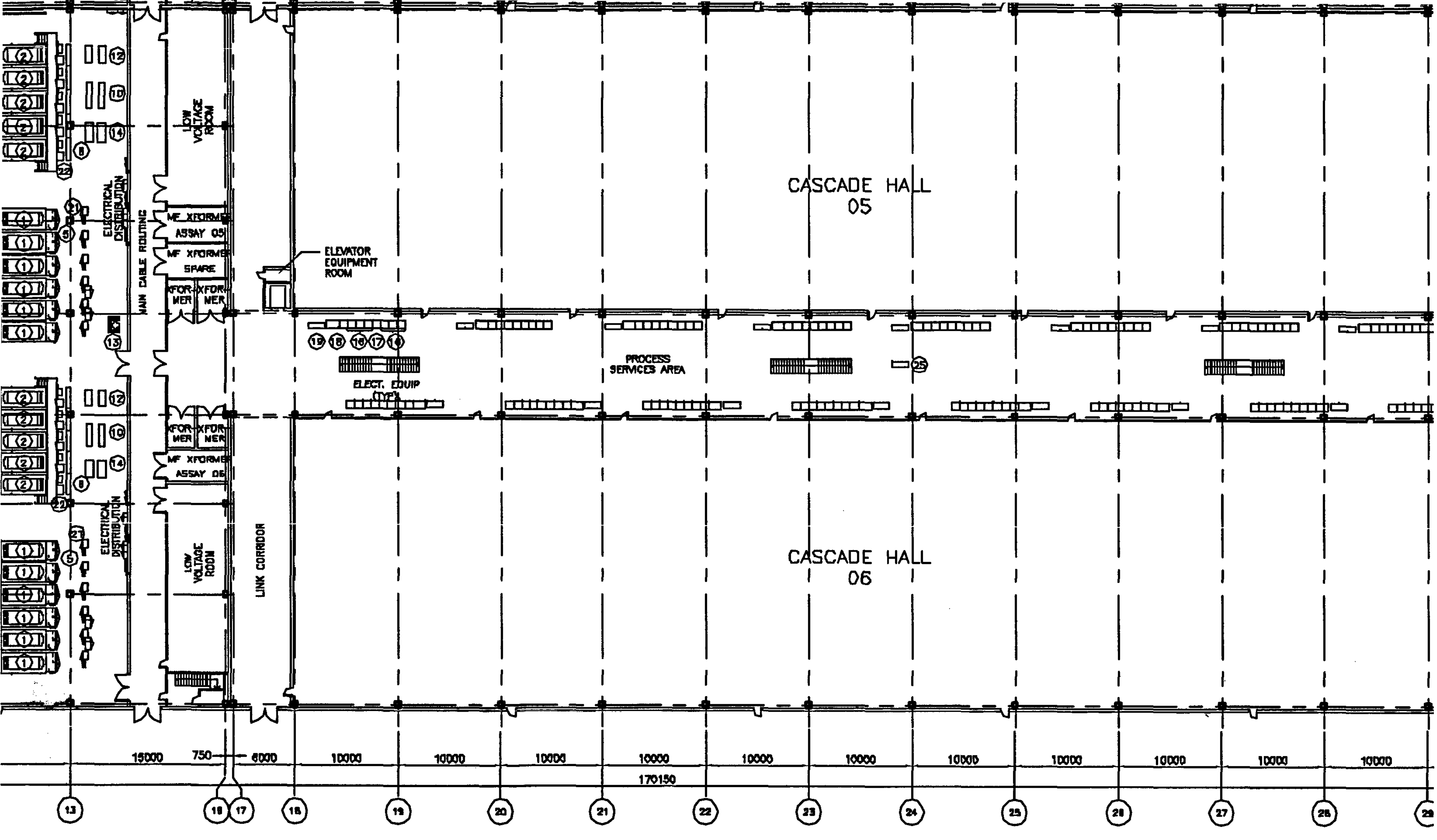
REFERENCE NUMBER
Figure_1.1-3.dwg

FIGURE 1.1-3
PLOT PLAN
(1 MILE RADIUS)

REVISION 2 DATE: JULY 2004

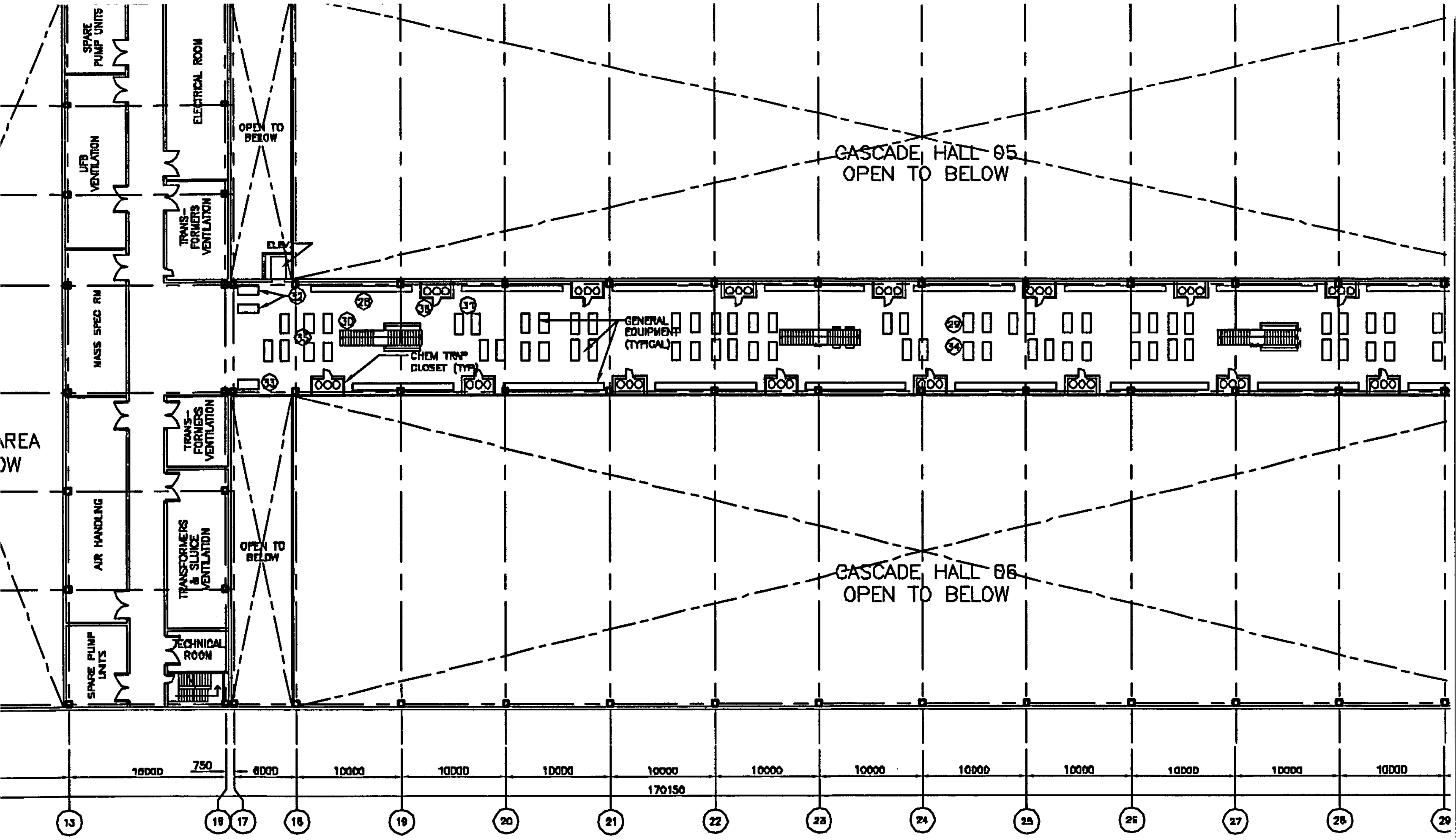


6019.52'
32
HIGH POINT OF DITCH
UBC STORAGE PAD
SITE BOUNDARY FENCE (TYP.)
3176.73'
EAST DIVERSION DITCH WITH 3:1 SIDE SLOPES
CONTROLLED ACCESS AREA BOUNDARY
FF ELEV. 3415
UBC STORAGE PAD STORMWATER RETENTION BASIN
TREATED EFFLUENT EVAPORATIVE BASIN
SITE STORMWATER DETENTION BASIN
33
2769.78'
5377.12'
NM HIGHWAY 234
NATURAL GAS LINE
OUTFALL OF DIVERSION DITCH
2840.43'
CO2 GAS LINE
ANDREWS, TX.
NEW MEXICO
TEXAS
NORTH
BOUNDAR
LEGEND:
1 VISITOR
2 SECURITY
3 ADMINIST
4 LIQUID N
5 CASCADE
6 CASCADE
7 CASCADE
8 ISO FREI
9 UBC STO
10 UF6 HAN
11 FIRE WAT
12 ELECTRIC
13 COOLING
14 CAB
15 CUB
16 TSB
17 CRDB
18 EMPLOYE
19 TRAILER
20 METEORL
21 BLENDING
22 GUARD H



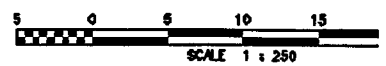
1,000,000 SWU MODULE

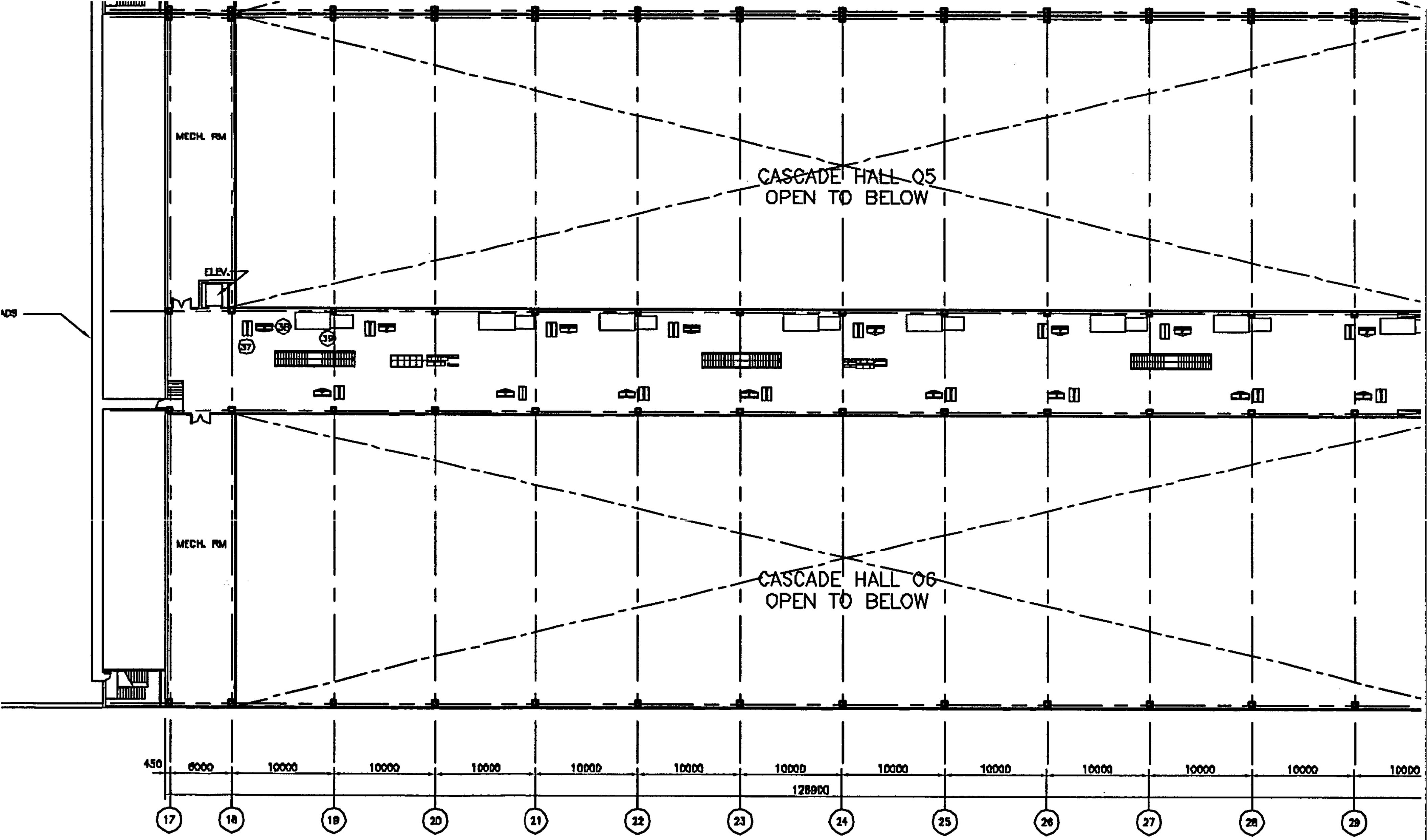
21 FEED STATION LOCAL CONTROL CENTER
22 PRODUCT STATION LOCAL CONTROL CENTER
23 TAILS STATION LOCAL CONTROL CENTER
24 FEED PURIFICATION STATION LOCAL CONTROL CENTER
25 PROCESS SERVICES AREA LOCAL CONTROL CENTER
26 ◆
27 RAIL TRANSPORTER
28 CASCADE VALVE STATION



1,000,000 SWU MODULE

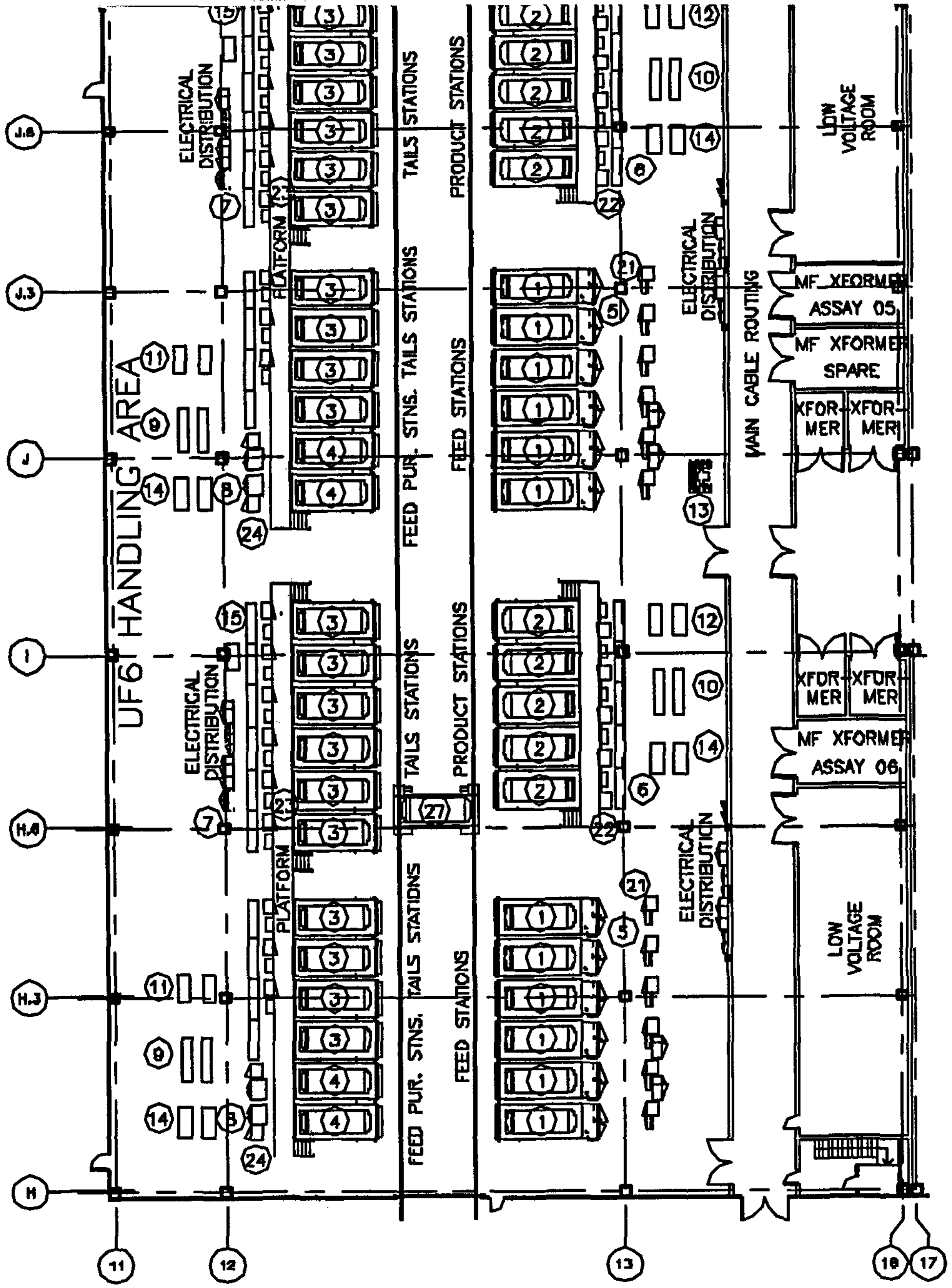
21 FEED STATION LOCAL CONTROL CENTER
22 PRODUCT STATION LOCAL CONTROL CENTER
23 TAILS STATION LOCAL CONTROL CENTER
24 FEED PURIFICATION STATION LOCAL CONTROL CENTER
25 PROCESS SERVICES AREA LOCAL CONTROL CENTER
26 *
27 RAIL TRANSPORTER
28 CASCADE VALVE STATION





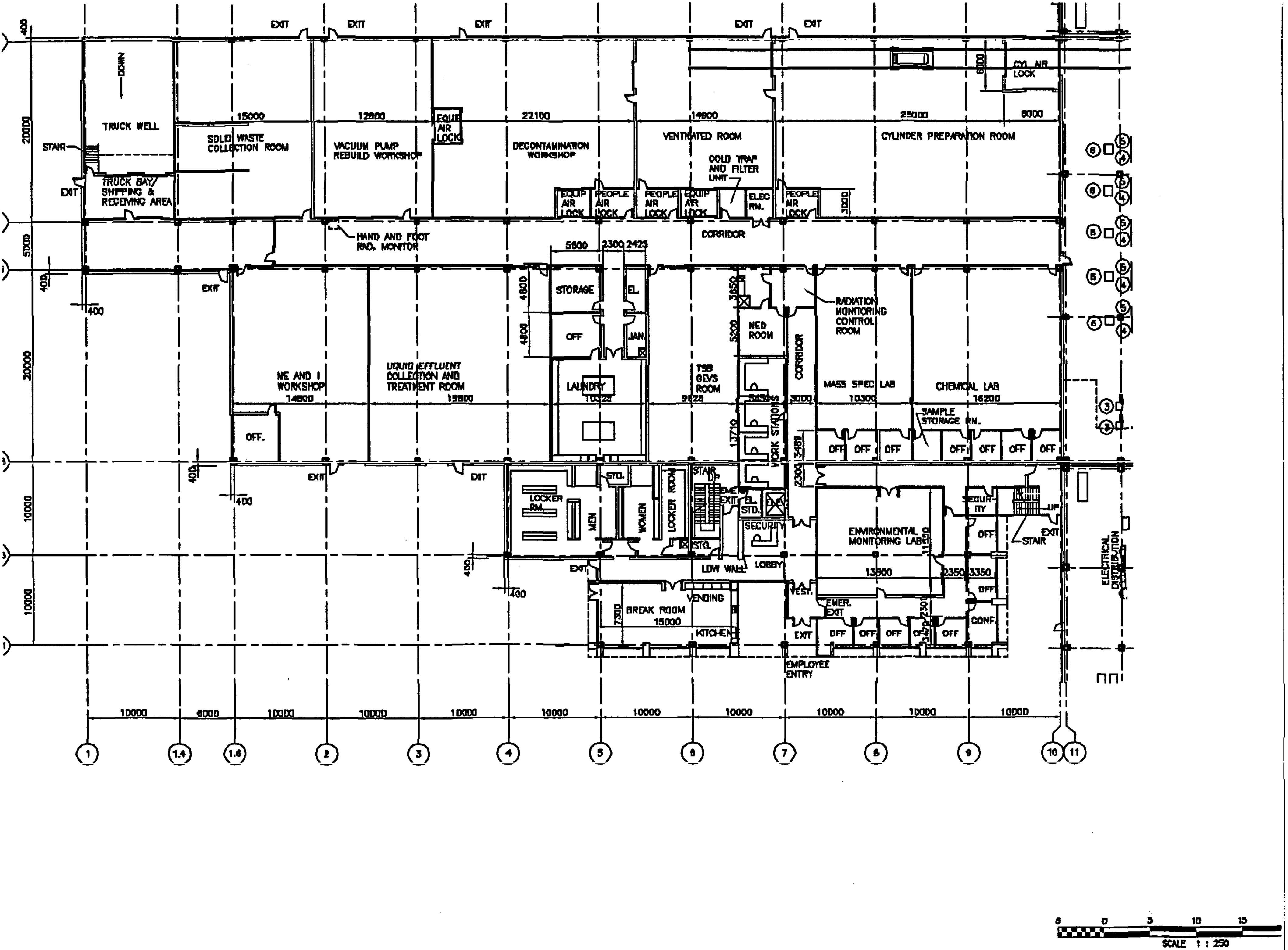
1,000,000 SWU MODULE

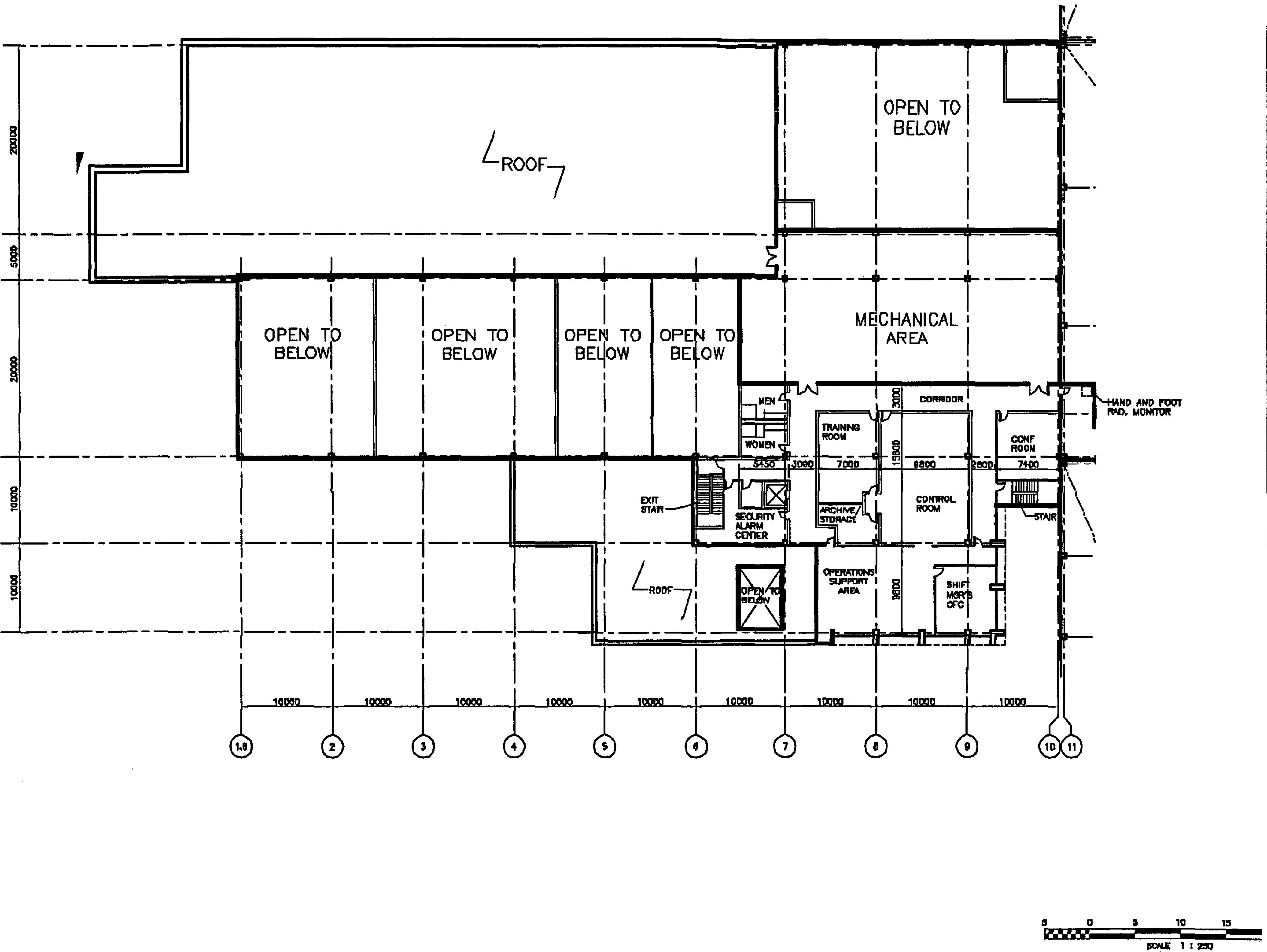
21 FEED STATION LOCAL CONTROL CENTER
22 PRODUCT STATION LOCAL CONTROL CENTER
23 TAILS STATION LOCAL CONTROL CENTER
24 FEED PURIFICATION STATION LOCAL CONTROL CENTER
25 PROCESS SERVICES AREA LOCAL CONTROL CENTER
26 *
27 RAIL TRANSPORTER
28 CASCADE VALVE STATION
29 PRODUCT PUMPING TRAIN

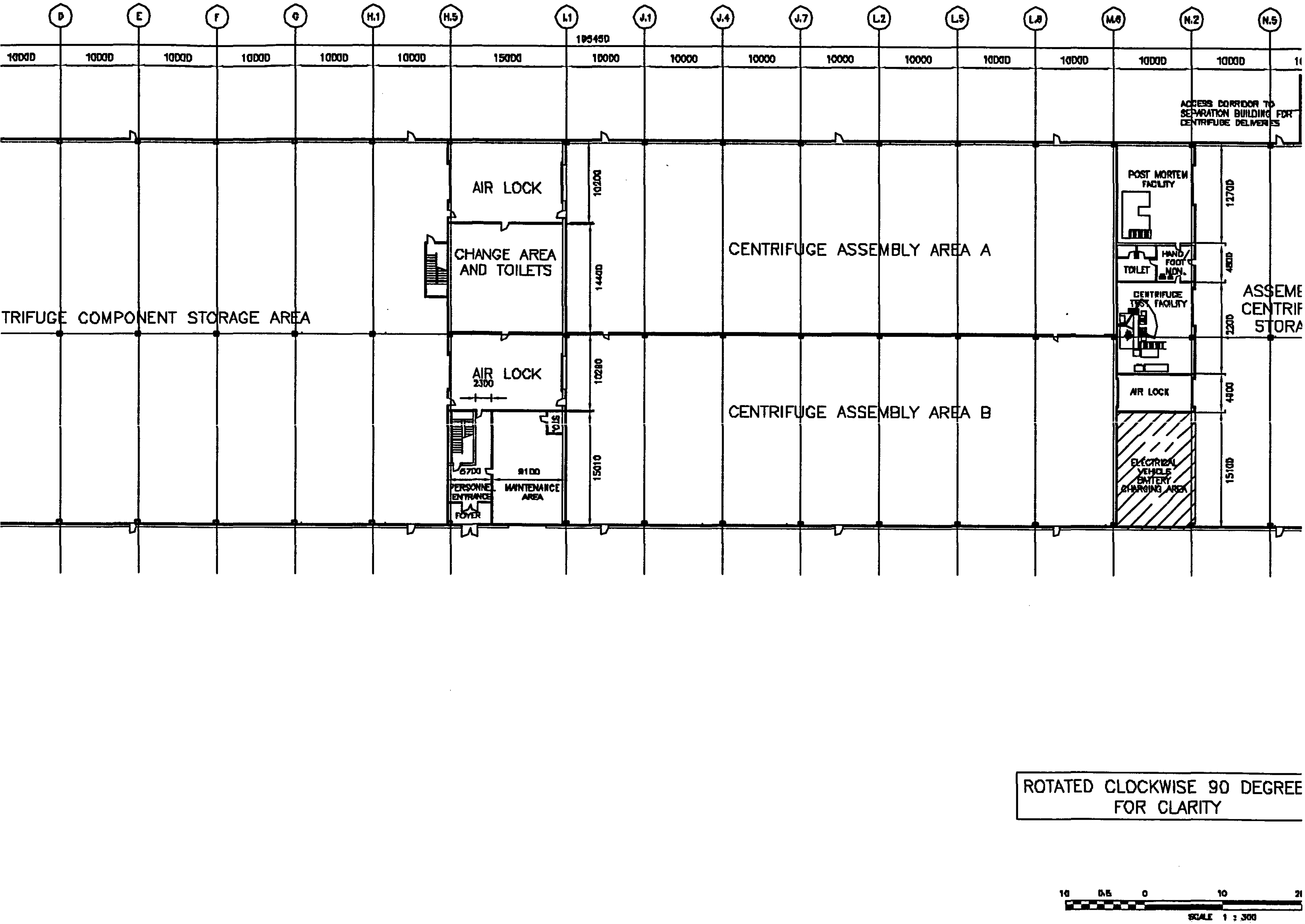


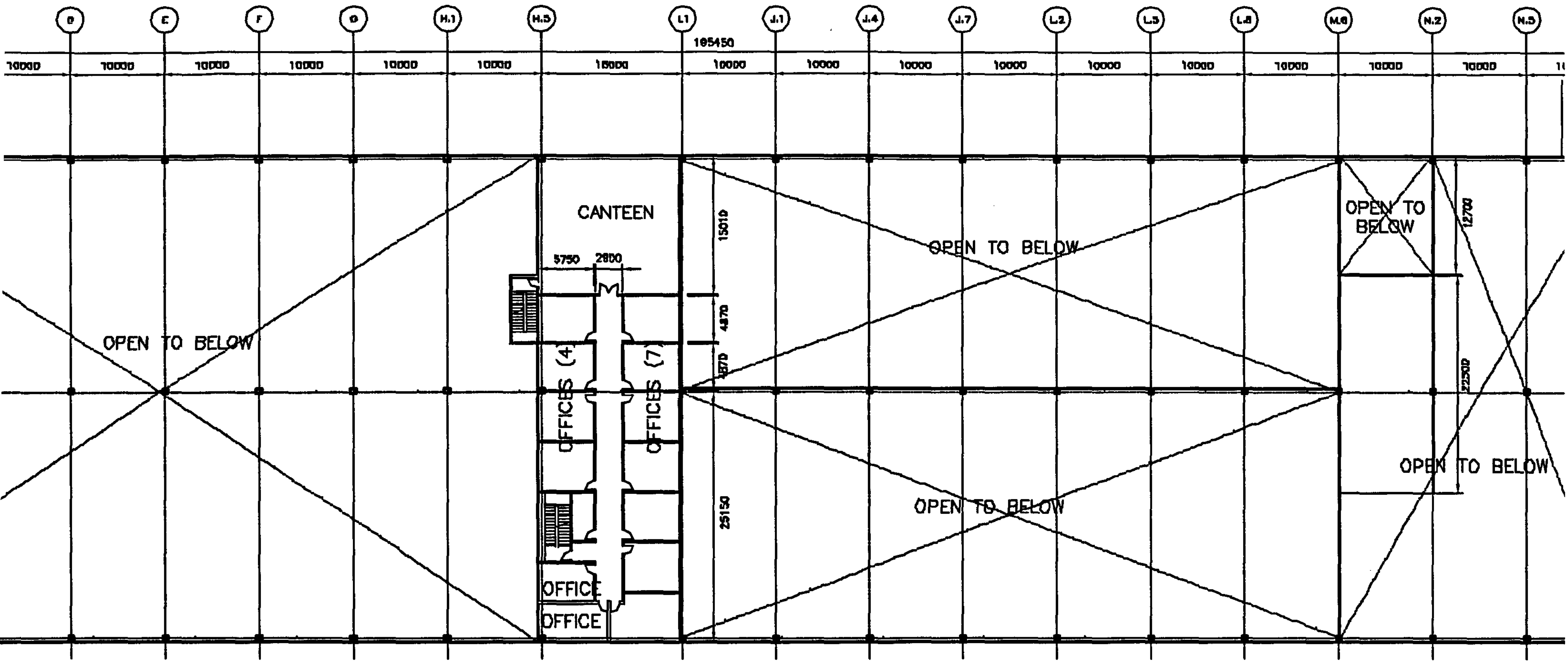
1,000,000 SWU MODULE

21 FEED STATION LOCAL CONTROL CENTER
22 PRODUCT STATION LOCAL CONTROL CENTER
23 TAILS STATION LOCAL CONTROL CENTER
24 FEED PURIFICATION STATION LOCAL CONTROL CENTER
25 PROCESS SERVICES AREA LOCAL CONTROL CENTER
26 *
27 RAIL TRANSPORTER
28 CASCADE VALVE STATION


EXIT
TRUCK WELL
DOWN
STAIR
TRUCK BAY/ SHIPPING & RECEIVING AREA
SOLID WASTE COLLECTION ROOM
VACUUM PUMP REBUILD WORKSHOP
EQUIP AIR LOCK
DECONTAMINATION WORKSHOP
VENTILATED ROOM
COLD TRAP AND FILTER UNIT
CYLINDER PREPARATION ROOM
CYL AIR LOCK
PEOPLE AIR LOCK
ELEC RM.
CORRIDOR
HAND AND FOOT RAD. MONITOR
STORAGE
EL.
OFF
JAN.
LAUNDRY
NE AND I WORKSHOP
LIQUID EFFLUENT COLLECTION AND TREATMENT ROOM
TSB GLVS ROOM
NED ROOM
RADIATION MONITORING CONTROL ROOM
MASS SPEC LAB
CHEMICAL LAB
SAMPLE STORAGE RM.
WORK STATIONS
OFF.
LOCKER RM.
MEN
WOMEN
LOCKER ROOM
STD.
STAIR
UP
EMER. EXIT
SECURITY
ENVIRONMENTAL MONITORING LAB
LOW WALL
LOBBY
BREAK ROOM
VENDING
KITCHEN
VEST.
CONF.
EMPLOYEE ENTRY
ELECTRICAL DISTRIBUTION
SCALE 1 : 250

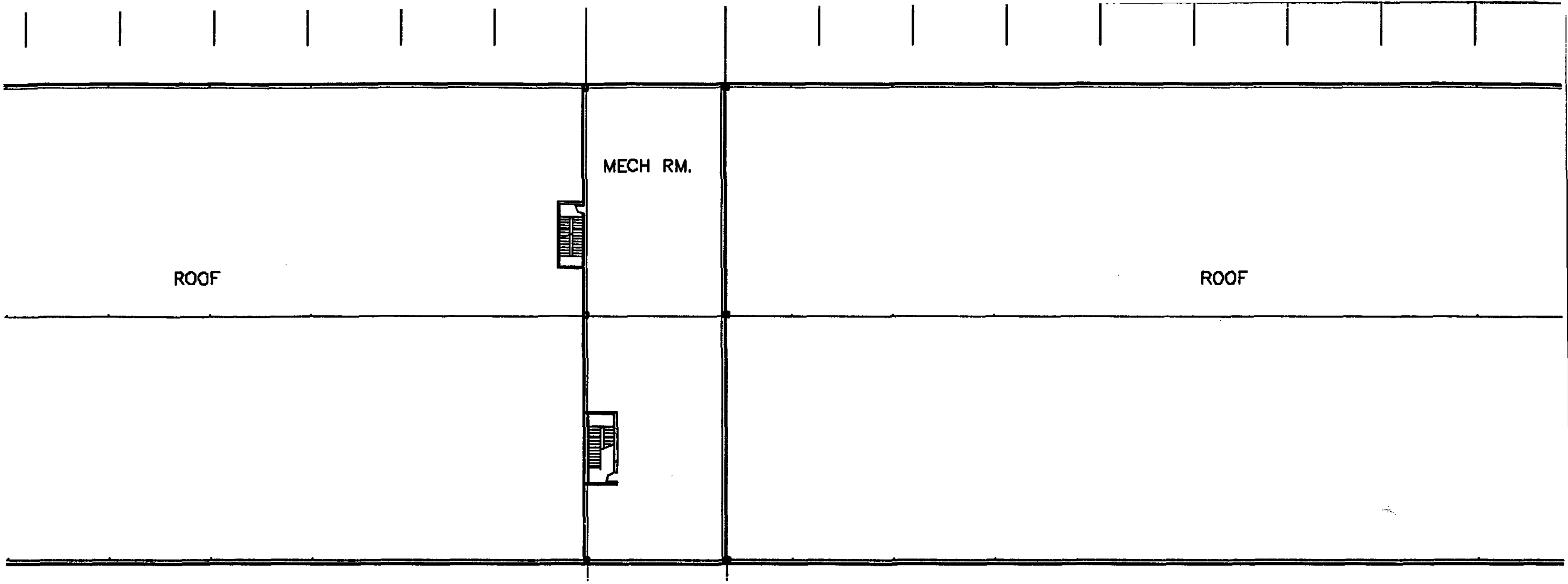

ROOF
OPEN TO BELOW
OPEN TO BELOW
OPEN TO BELOW
OPEN TO BELOW
OPEN TO BELOW
OPEN TO BELOW
MECHANICAL AREA
MEN
WOMEN
CORRIDOR
TRAINING ROOM
CONF ROOM
HAND AND FOOT RAD. MONITOR
EXIT STAIR
SECURITY ALARM CENTER
ARCHIVE/ STORAGE
CONTROL ROOM
STAIR
OPERATIONS SUPPORT AREA
SHIFT MGR'S OFC
ROOF
5450
3000
7000
9800
2800
7400
3000
19800
9800
20000
5000
20000
10000
10000
10000
10000
10000
10000
10000
10000
10000
10000
10000
1.8
2
3
4
5
6
7
8
9
10
11
5
0
5
10
15
SCALE 1 : 250


D
E
F
G
H.1
H.5
I.1
J.1
J.4
J.7
L.2
L.5
L.8
M.6
N.2
N.5
185450
10000
15000
ACCESS CORRIDOR TO SEPARATION BUILDING FOR CENTRIFUGE DELIVERIES
AIR LOCK
CHANGE AREA AND TOILETS
AIR LOCK
2300
CENTRIFUGE ASSEMBLY AREA A
CENTRIFUGE ASSEMBLY AREA B
TRIFUGE COMPONENT STORAGE AREA
6700
9100
PERSONNEL ENTRANCE
FOYER
MAINTENANCE AREA
10200
14400
10280
15010
POST MORTEM FACILITY
TOILET
HAND/ FOOT MON.
CENTRIFUGE TEST FACILITY
AIR LOCK
ELECTRICAL VEHICLE BATTERY CHARGING AREA
12700
4800
2200
4400
15100
ASSEME
CENTRIF
STORA
ROTATED CLOCKWISE 90 DEGREE FOR CLARITY
SCALE 1 : 300

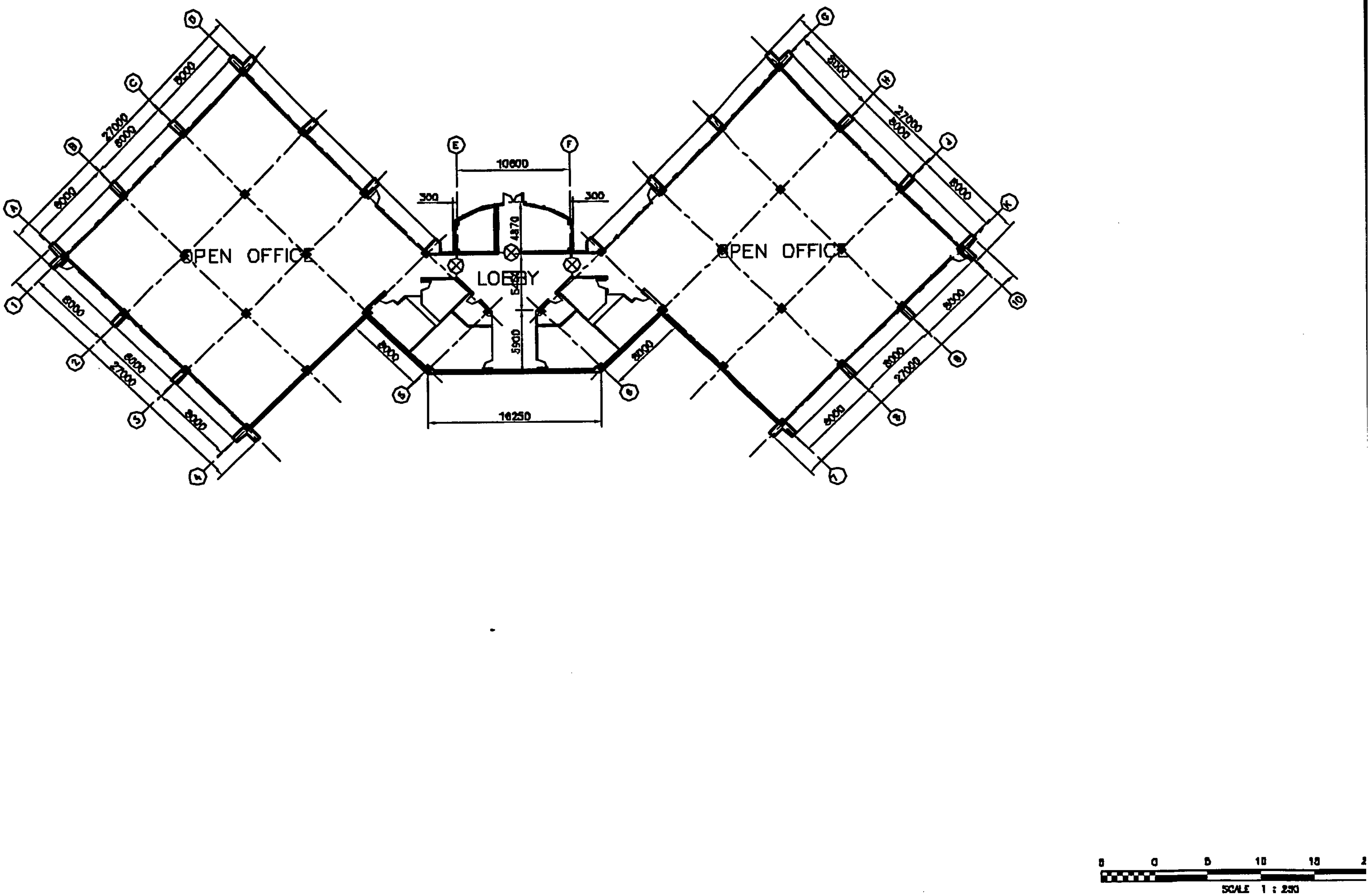


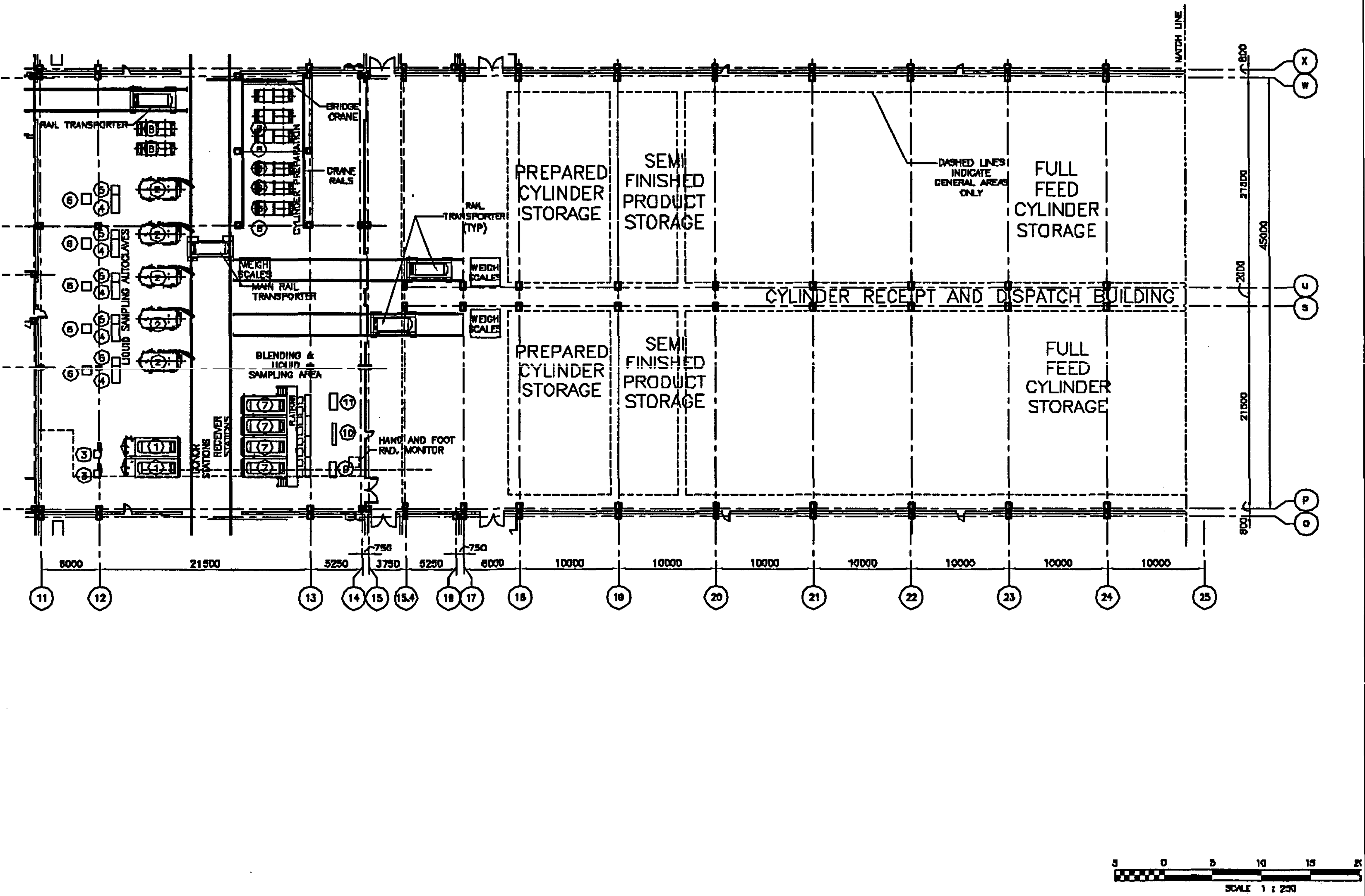
ROTATED CLOCKWISE 90 DEGREE
FOR CLARITY

SCALE 1 : 300

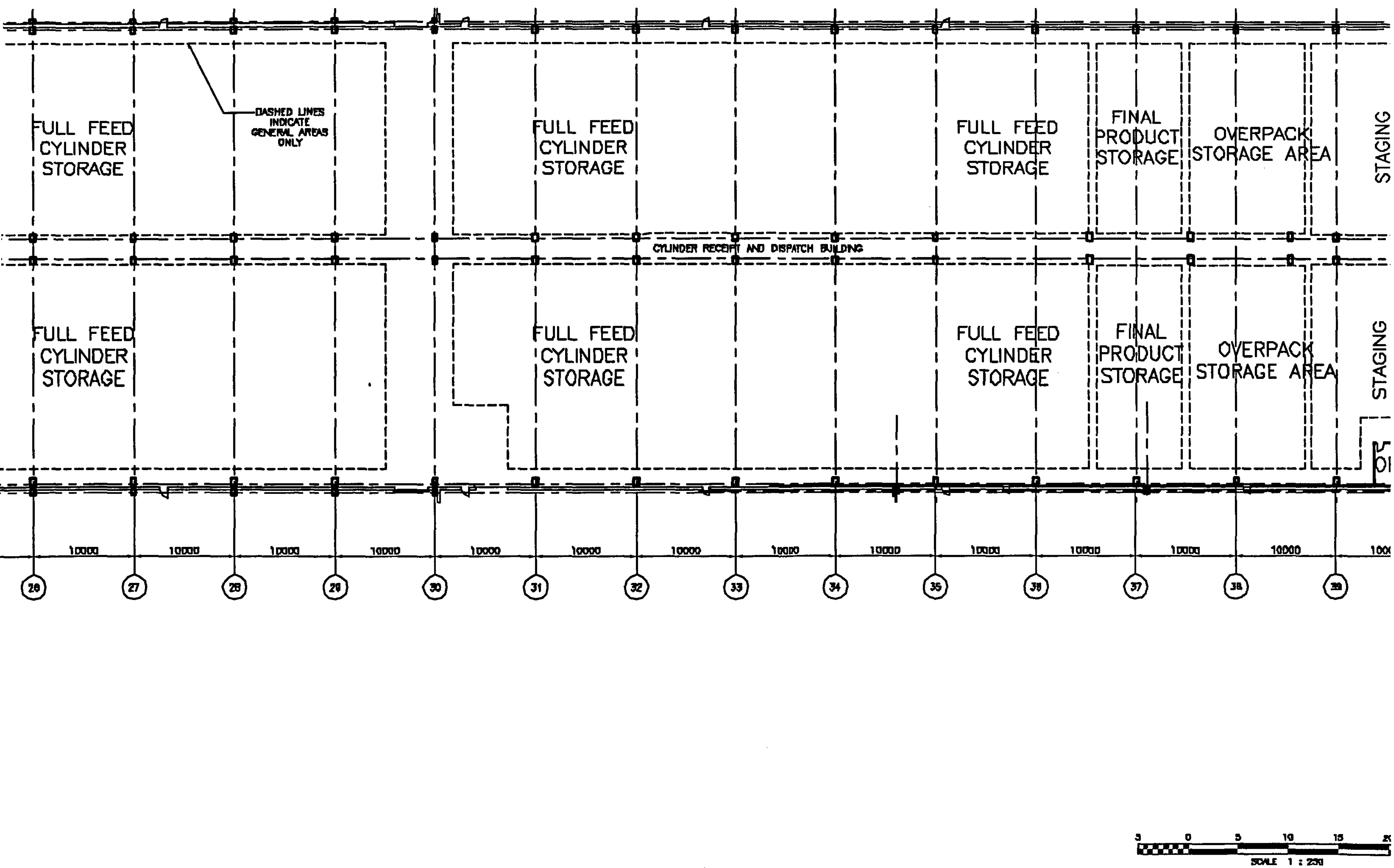


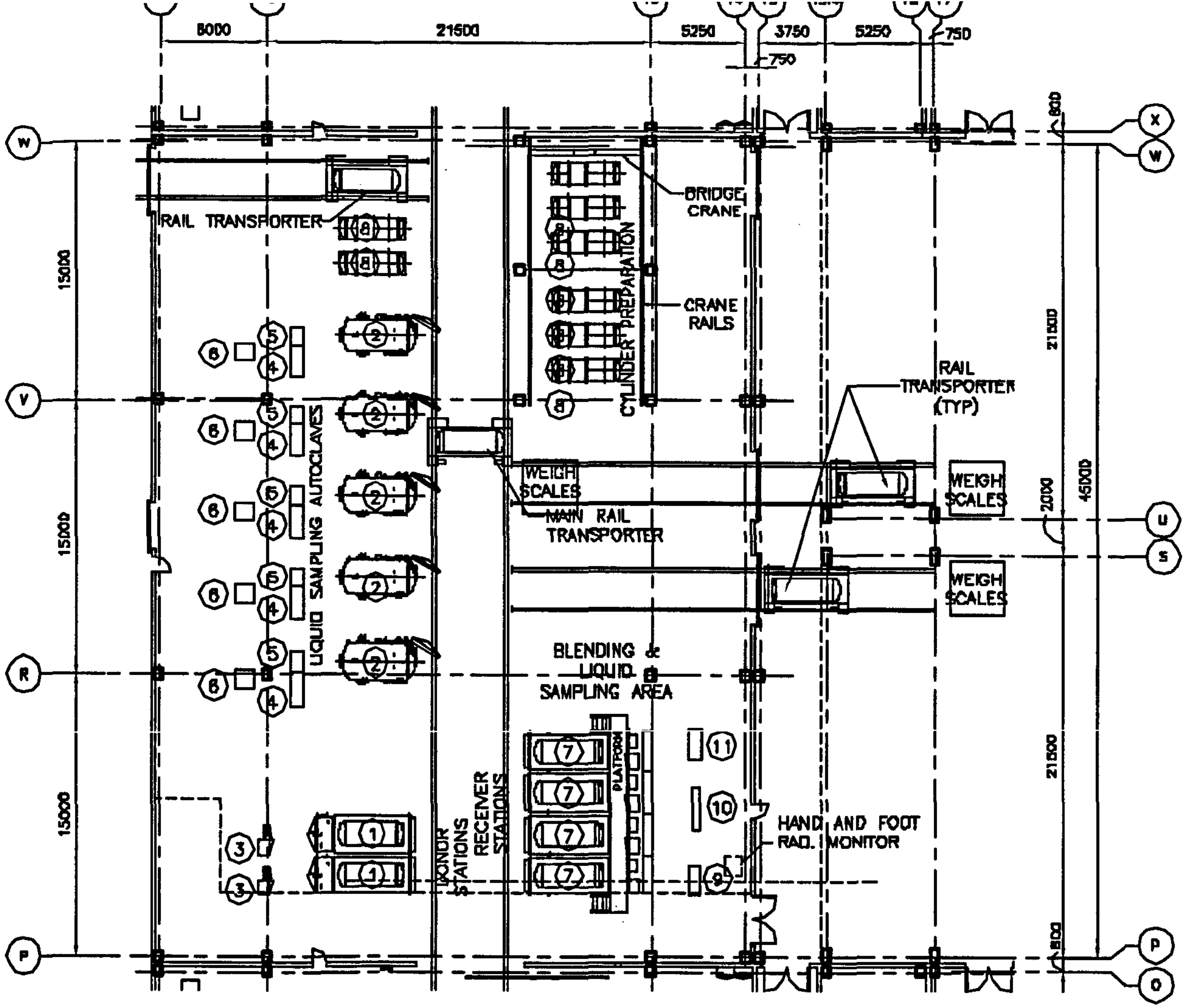
ROTATED CLOCKWISE 90 DEGREE
FOR CLARITY


OPEN OFFICE
LOBBY
OPEN OFFICE
10600
300
300
16250
27000
8000
SCALE 1 : 250

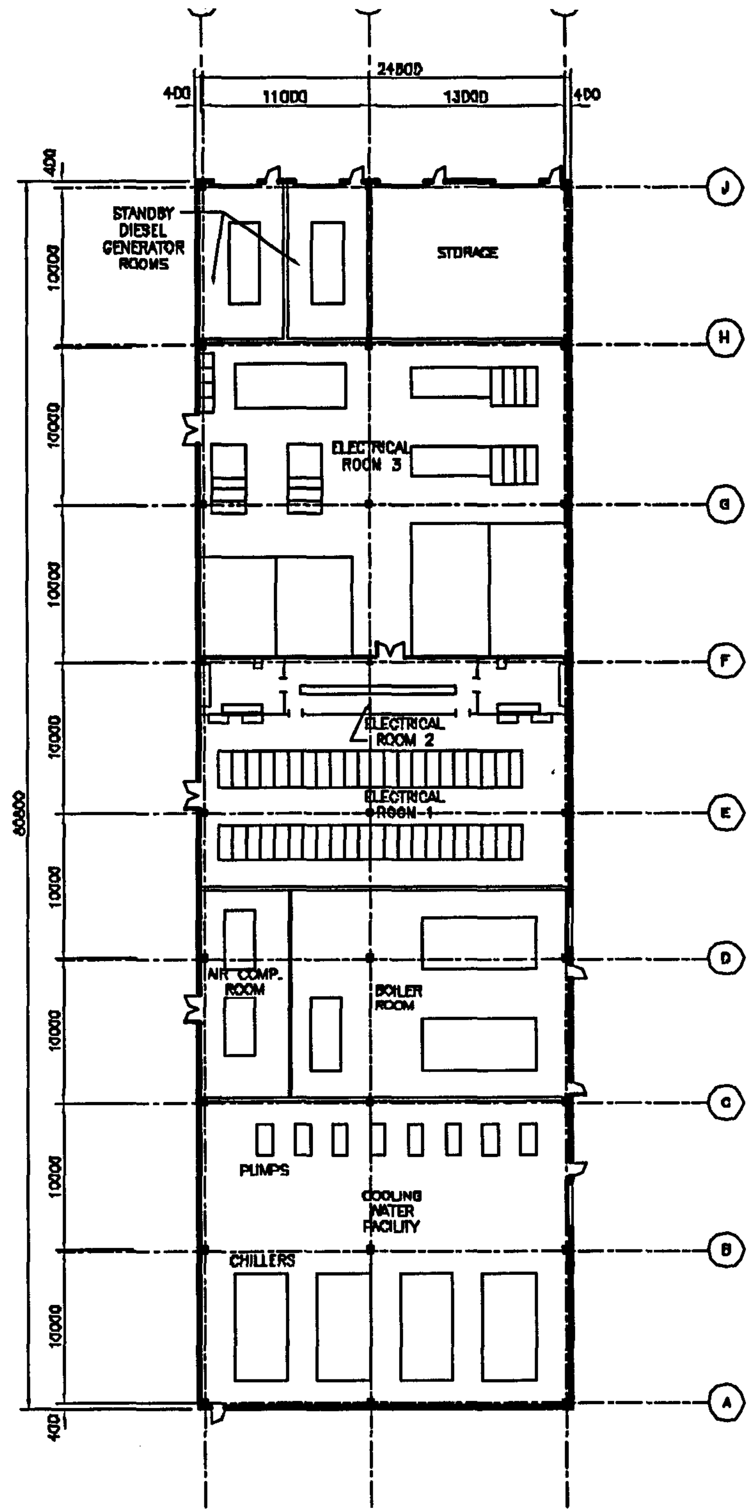


RAIL TRANSPORTER
BRIDGE CRANE
CRANE RAILS
CYLINDER PREPARATION
LIQUID SAMPLING AUTOCLAVES
WEIGH SCALES
MAIN RAIL TRANSPORTER
RAIL TRANSPORTER (TYP)
WEIGH SCALES
WEIGH SCALES
BLENDING & LIQUID SAMPLING AREA
PLATFORM
DONOR STATIONS
RECEIVER STATIONS
HAND AND FOOT RAD. MONITOR
PREPARED CYLINDER STORAGE
SEMI FINISHED PRODUCT STORAGE
DASHED LINES INDICATE GENERAL AREAS ONLY
FULL FEED CYLINDER STORAGE
CYLINDER RECEIPT AND DISPATCH BUILDING
PREPARED CYLINDER STORAGE
SEMI FINISHED PRODUCT STORAGE
FULL FEED CYLINDER STORAGE
MATCH LINE
800
21700
45000
2000
21500
800
X
W
U
S
P
O
750
750
6000
21500
5250
3750
5250
6000
10000
10000
10000
10000
10000
10000
10000
11
12
13
14
15
15.4
16
17
18
19
20
21
22
23
24
25
5
0
5
10
15
20
SCALE 1 : 250

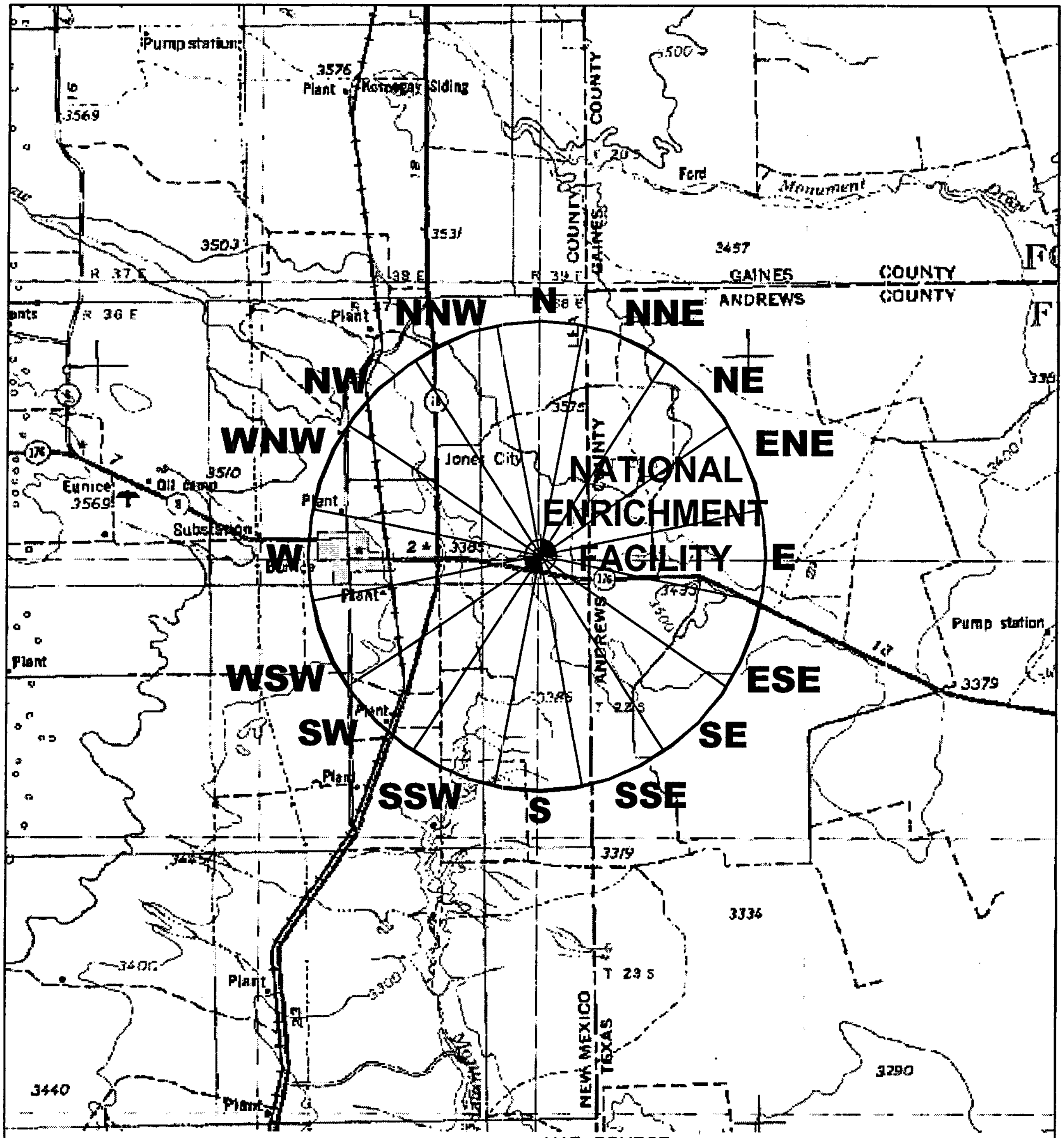

FULL FEED CYLINDER STORAGE
DASHED LINES INDICATE GENERAL AREAS ONLY
FULL FEED CYLINDER STORAGE
FULL FEED CYLINDER STORAGE
FINAL PRODUCT STORAGE
OVERPACK STORAGE AREA
STAGING
CYLINDER RECEIPT AND DISPATCH BUILDING
FULL FEED CYLINDER STORAGE
FULL FEED CYLINDER STORAGE
FULL FEED CYLINDER STORAGE
FINAL PRODUCT STORAGE
OVERPACK STORAGE AREA
STAGING
10000
26
27
28
29
30
31
32
33
34
35
36
37
38
39
SCALE 1 : 200



6000
21500
5250
3750
5250
750
750
RAIL TRANSPORTER
BRIDGE CRANE
CRANE RAILS
CYLINDER PREPARATION
LIQUID SAMPLING AUTOCLAVES
RAIL TRANSPORTER (TYP)
WEIGH SCALES
MAIN RAIL TRANSPORTER
WEIGH SCALES
WEIGH SCALES
BLENDING & LIQUID SAMPLING AREA
DONOR STATIONS
RECEIVER STATIONS
PLATFORM
HAND AND FOOT RAD. MONITOR
15000
15000
15000
21500
2000
45000
21500
SCALE 1 : 200


24800
400
11000
13000
400
400
10000
10000
10000
10000
10000
10000
10000
10000
80800
400
STANDBY DIESEL GENERATOR ROOMS
STORAGE
ELECTRICAL ROOM 3
ELECTRICAL ROOM 2
ELECTRICAL ROOM 1
AIR COMP. ROOM
BOILER ROOM
PUMPS
COOLING WATER FACILITY
CHILLERS
J
H
G
F
E
D
C
B
A


5
0
5
10
15
20
SCALE 1 : 250



1 0 1 2 3 4 5 6 7 8 9 10 KM

1 0 1 2 3 4 5 6 MILES

NORTH

CONTOUR INTERVAL: 100 FT

MAP SOURCE:
USGS 1:100,000 30 MIN. X 60 MIN. MAPS
HOBBS, NEW MEXICO – TEXAS
AND EUNICE NE,TEX. – N. MEX.

REFERENCE NUMBER
15Min Figures.dwg

FIGURE 1.3-1

RADIAL SECTORS
(5 MILE RADIUS)

REVISION DATE: DECEMBER 2003

TABLE OF CONTENTS

LIST OF FIGURES

2.0 ORGANIZATION AND ADMINISTRATION

This chapter describes the management system and administrative procedures for the effective implementation of Health, Safety, and Environmental (HS&E) functions at the Louisiana Energy Services (LES) enrichment facility. The chapter presents the organizations responsible for managing the design, construction, operation, and decommissioning of the facility. The key management and supervisory positions and functions are described including the personnel qualifications for each key position at the facility.

The facility organization, technical qualifications, procedures, and management controls in this license application are similar to those submitted for Nuclear Regulatory Commission (NRC) review in the LES license application for the Claiborne Enrichment Center (LES, 1993). The NRC staff evaluated the proposed organization and administrative procedures and concluded that they were adequate and met the requirements specified in 10 CFR 40 (CFR, 2003b) and 70 (CFR, 2003c) concerning organizational structure; staff technical qualifications, functions, and responsibilities; and management controls (NRC, 1994). LES has modified the facility operating organization from the one previously accepted to better reflect lessons learned and operating experience at Uranium Enrichment Company (Urenco) facilities and United States nuclear facilities. Although some position titles and scope of responsibility have been changed, the functions to be performed by the operating organization remain the same as the Claiborne Enrichment Center submittal.

The LES policy is to maintain a safe work place for its employees and to assure operational compliance within the terms and conditions of the license and applicable regulations. The Plant Manager has overall responsibility for safety and compliance to this policy. In particular, LES employs the principle of keeping radiation and chemical exposures to employees and the general public as low as reasonably achievable (ALARA).

The information provided in this chapter, the corresponding regulatory requirement, and the section of NUREG-1520 (NRC, 2002), Chapter 2 in which the NRC acceptance criteria are presented is summarized below.

Information Category and Requirement	10 CFR 70 Citation	NUREG-1520 Chapter 2 Reference
Section 2.1 Organizational Structure		
• Functional description of specific organization groups responsible for managing the design, construction, and operation of the facility	70.22(a)(6)	2.4.3(1) & 2.4.3(7)
• Management controls and communications among organizational units	70.22(a)(8)	2.4.3(2)
• Startup and transition to operations	70.22(a)(6)	2.4.3(4)
Section 2.2 Key Management Positions		
• Qualifications, responsibilities, and authorities for key management personnel	70.22(a)(6)	2.4.3(3) & 2.4.3(4)
Section 2.3 Administration		
• Effective implementation of HS&E functions using written procedures	70.22(a)(8)	2.4.3(5)
• Reporting of unsafe conditions or activities	70.62(a)	2.4.3(6)
• Commitment to establish formal management measures to ensure availability of IROFS	70.62(d)	2.4.3(8)
• Written agreements with offsite emergency resources	70.22(i)	2.4.3(9)

2.1 ORGANIZATIONAL STRUCTURE

The LES organizational structure is described in the following sections. The organizational structure indicates the lines of communication and management control of activities associated with the design, construction, operation, and decommissioning of the facility.

2.1.1 Corporate Functions, Responsibilities, and Authorities

LES is a registered limited partnership formed solely to provide uranium enrichment services for commercial nuclear power plants. The LES partnership is described in Chapter 1, Section 1.2, Institutional Information.

LES has presented to Lea County, New Mexico a proposal to develop the NEF. Lea County would issue its Industrial Revenue Bond (National Enrichment Facility Project) Series 2004 in the maximum aggregate principal amount of $1,800,000,000 to accomplish the acquisition, construction and installation of the project pursuant to the County Industrial Revenue Bond Act, Chapter 4, Article 59 NMSA 1978 Compilation, as amended. The Project is comprised of the land, buildings, and equipment.

Under the Act, Lea County is authorized to acquire industrial revenue projects to be located within Lea County but outside the boundaries of any incorporated municipality for the purpose of promoting industry and trade by inducing manufacturing, industrial and commercial enterprises to locate or expand in the State of New Mexico, and for promoting a sound and proper balance in the State of New Mexico between agriculture, commerce, and industry. Lea County will lease the project to LES, and LES will be responsible for the construction and operation of the facility. Upon expiration of the Bond after 30 years, LES will purchase the project.

The County has no power under the Act to operate the project as a business or otherwise or to use or acquire the project property for any purpose, except as lessor thereof under the terms of the lease.

In the exercise of any remedies provided in the lease, the County shall not take any action at law or in equity that could result in the Issuer obtaining possession of the project property or operating the project as a business or otherwise.

LES is responsible for the design, quality assurance, construction, operation, and decommissioning of the enrichment facility. The President of LES reports to the LES Management Committee. This committee is composed of representatives from the general partners of LES.

The President receives policy direction from the LES Management Committee. Reporting to the President are the Engineering and Contracts Manager, the Corporate Communications Manager, Chief Financial Officer (CFO), the Quality Assurance (QA) Director, Chief Operating Officer (COO), and the Health, Safety & Environment Director. Figure 2.1-1, LES Corporate, Design and Construction Organization shows the authority and lines of communication.

2.1.2 Design and Construction Organization

As the owner of the enrichment technology and operator of the enrichment facilities in Europe, LES has contracted Urenco Limited to prepare the reference design for the facility, while an architect/engineering (A/E) has been contracted to further specify structures and systems of the facility, and ensure the reference design meets all applicable U.S. codes and standards. A contractor specializing in site evaluations has been contracted to perform the site selection evaluation. A nuclear consulting company has been contracted to conduct the site characterization, perform the Integrated Safety Analysis and to support development of the license application.

During the construction phase, preparation of construction documents and construction itself are contracted to qualified contractors. The Engineering and Contracts Manager is responsible for managing the design, construction, initial startup and procurement activities. Contractor QA Programs will be reviewed by LES QA and must be approved before work can start.

Urenco will design, manufacture and deliver to the site the centrifuges necessary for facility operation. In addition, Urenco is supplying technical assistance and consultation for the facility. Urenco has extensive experience in the gas centrifuge uranium enrichment process since it operates three gas centrifuge uranium enrichment plants in Europe. Urenco is conducting technical reviews of the design activities to ensure the design of the enrichment facility is in accordance with the Urenco reference design information.

For procurement involving the use of vendors located outside the U.S., LES selects vendors only after a determination that their quality assurance programs meet the LES requirements. Any components supplied to LES are designed to meet applicable domestic industry code requirements or their equivalents as stated by the equipment specifications.

As shown in Figure 2.1-1, the Engineering and Contracts Manager is responsible for managing the work and contracts with the Technology Supplier (Urenco) and a select group of Project Managers. These Project Managers will be responsible for the areas of Procurement, Construction, Engineering, Project Engineering, Project Controls and Start-up. The lines of communication of key management positions within the engineering and construction organization are shown in Figure 2.1-1.

Position descriptions of key management personnel in the design and construction organization will be accessible to all affected personnel and the NRC.

2.1.3 Operating Organization

The operating organization for LES is shown in Figures 2.1-1, and 2.1-2, LES National Enrichment Facility Operating Organization. LES has direct responsibility for preoperational testing, initial start-up, operation and maintenance of the facility.

The Plant Manager reports to the COO and is responsible for the overall operation and administration of the enrichment facility. He is also responsible for ensuring the facility complies with all applicable regulatory requirements. In the discharge of these responsibilities, he directs the activities of the following groups:

- Health, Safety, and Environment

- Operations
- Uranium Management
- Technical Services
- Human Resources
- Quality Assurance.

The responsibilities, authorities and lines of communication of key management positions within the operating organization are discussed in Section 2.2, Key Management Positions.

During the Operations Phase the QA Manager reports to the Plant Manager. However, the QA Manager has the authority and responsibility to contact directly the LES President, through the QA Director, with any Quality Assurance concerns during operation.

Position descriptions for key management personnel in the operating organization will be accessible to all affected personnel and to the NRC.

2.1.4 Transition From Design and Construction to Operations

LES is responsible for the design, quality assurance, construction, testing, initial startup, operation, and decommissioning of the facility.

Towards the end of construction, the focus of the organization will shift from design and construction to initial start-up and operation of the facility. As the facility nears completion, LES will staff the LES NEF Operating Organization to ensure smooth transition from construction activities to operation activities. During this transition, the Health, Safety, & Environment (HS&E) Manager position reports directly to the LES President (as shown in Figure 2.1-1) for HS&E matters related to design and construction and reports directly to the Plant Manager (as shown in Figure 2.1-2) for HS&E matters related to operations. This position is intentionally duplicated to provide significant continued focus on the health, safety, and environment goals during design and construction when the operating organization is not yet fully developed and implemented. Urenco, which has been operating gas centrifuge enrichment facilities in Europe for over 30 years, will have personnel integrated into the LES organization to provide technical support during startup of the facility and transition into the operations phase.

As the construction of systems is completed, the systems will undergo acceptance testing as required by procedure, followed by turnover from the construction organization to the operations organization by means of a detailed transition plan. The turnover will include the physical systems and corresponding design information and records. Following turnover, the operating organization will be responsible for system maintenance and configuration management. The design basis for the facility is maintained during the transition from construction to operations through the configuration management system described in Chapter 11, Management Measures.

Additional information regarding the transition from design and construction to operations, for the LES QA Organization, is provided in Section 1 of the LES Quality Assurance Program Description (i.e., Appendix A of the NEF Safety Analysis Report).

2.2 KEY MANAGEMENT POSITIONS

This section describes the functional positions responsible for managing the operation of the facility. The facility is staffed at sufficient levels prior to operation to allow for training, procedure development, and other pre-operational activities.

The responsibilities, authorities and lines of communication for each key management position are provided in this section. Responsible managers have the authority to delegate tasks to other individuals; however, the responsible manager retains the ultimate responsibility and accountability for implementing the applicable requirements. Management responsibilities, supervisory responsibilities, and the criticality safety engineering staff responsibilities related to nuclear criticality safety are in accordance with ANSI/ANS-8.19-1996, Administrative Practices for Nuclear Criticality Safety (ANSI, 1996).

The LES Corporate Organization and lines of communication are shown in Figure 2.1-1.

2.2.1 Operating Organization

The functions and responsibilities of key facility management are described in the following paragraphs. Additional detailed responsibilities related to nuclear criticality safety for key management positions and remaining supervisory and criticality safety staff are in accordance with ANSI/ANS-8.19-1996 (ANSI, 1996). The basic functions and responsibilities are the same as that previously accepted by the NRC Staff in NUREG-1491, Section 10 (NRC, 1994). Some position titles have been changed to better reflect the actual responsibilities of the position. Similarly, some operating functions have been assigned to different managers to better reflect the operating organization presently used at Urenco and U. S. nuclear facilities.

A. Chief Operating Officer

The Chief Operating Officer (COO) is appointed by the President and is responsible for ensuring the facility complies with all applicable regulatory requirements. The COO directs these responsibilities through the Plant Manager.

B. Plant Manager

The Plant Manager shall be appointed by, and report to, the Chief Operating Officer of LES. The Plant Manager has direct responsibility for operation of the facility in a safe, reliable and efficient manner. The Plant Manager is responsible for proper selection of staff for all key positions including positions on the Safety Review Committee. The Plant Manager is responsible for the protection of the facility staff and the general public from radiation and chemical exposure and/or any other consequences of an accident at the facility and also bears the responsibility for compliance with the facility license. The Plant Manager or designee(s) have the authority to approve and issue procedures.

C. Quality Assurance Director

The Quality Assurance Director is appointed by and reports to the President and has overall responsibility for development, management and implementation of the LES QA Program.

D. Quality Assurance Manager

The Quality Assurance (QA) Manager reports to the Plant Manager and is responsible for establishing and maintaining the Quality Assurance Program for the facility. The facility line managers and their staff who are responsible for performing quality-affecting work are responsible for ensuring implementation of and compliance with the QA Program. The QA Manager position is independent from other management positions at the facility to ensure the QA Manager has access to the Plant Manager for matters affecting quality. In addition, the QA Manager has the authority and responsibility to contact the LES President through the QA Director with any Quality Assurance concerns.

E. Health, Safety, and Environment Manager

The Health, Safety, and Environment (HS&E) Manager reports to the Plant Manager and has the responsibility for assuring safety at the facility through activities including maintaining compliance with safeguards, appropriate rules, regulations, and codes and has responsibility for implementation and control of the Fundamental Nuclear Material Control (FNMC) Plan. This includes HS&E activities associated with nuclear criticality safety, radiation protection, chemical safety, environmental protection, emergency preparedness and industrial safety. The HS&E Manager works with the other facility managers to ensure consistent interpretations of HS&E requirements, performs independent reviews, and supports facility and operations change control reviews.

This position is independent from other management positions at the facility to ensure objective HS&E audit, review, and control activities. The HS&E Manager has the authority to shut down operations if they appear to be unsafe, and must consult with the Plant Manager with respect to restart of shutdown operations after the deficiency, or unsatisfactory condition, has been resolved.

F. Operations Manager

The Operations Manager reports to the Plant Manager and has the responsibility of directing the day-to-day operation of the facility. This includes such activities as ensuring the correct and safe operation of UF_6 processes, proper handling of UF_6, and the identification and mitigation of any off normal operating conditions. In the event of the absence of the Plant Manager, the Operations Manager may assume the responsibilities and authorities of the Plant Manager.

G. Uranium Management Manager

The Uranium Management Manager reports to the Plant Manager and has the responsibility for UF_6 cylinder management (including transportation licensing) and directing the scheduling of enrichment operations to ensure smooth production. This includes activities such as ensuring proper feed material and maintenance equipment are available for the facility. In the event of the absence of the Plant Manager, the Uranium Management Manager may assume the responsibilities and authorities of the Plant Manager.

H. Technical Services Manager

The Technical Services Manager reports to the Plant Manager and has the responsibility of providing technical support to the facility. This includes technical support for facility modifications (including administration of the configuration management system), engineering support for operations and maintenance, performance, operation of the chemistry laboratory, maintenance activities, and computer support. In the event of the absence of the Plant Manager, the Technical Services Manager may assume the responsibilities and authorities of the Plant Manager.

I. Human Resources Manager

The Human Resource Manager reports to the Plant Manager and has the responsibility for community relations, ensuring adequate staffing, ensuring training is provided for facility employees, providing administrative support services to the facility including document control, and for the physical security of the facility.

J. Quality Assurance Inspectors

The Quality Assurance Inspectors report to the Quality Assurance Manager (via a designated supervisory position, if applicable) and have the responsibility for performing inspections related to the implementation of the LES QA Program.

K. Quality Assurance Auditors

The Quality Assurance Auditors report to the Quality Assurance Manager (via a designated supervisory position, if applicable) and have the responsibility for performing audits related to the implementation of the LES QA Program.

L. Quality Assurance Technical Support

The Quality Assurance Technical Support personnel report to the Quality Assurance Manager (via a designated supervisory position, if applicable) and have the responsibility for providing technical support related to the implementation of the LES QA Program.

M. Emergency Preparedness Manager

The Emergency Preparedness Manager reports to the HS&E Manager and has the responsibility for ensuring the facility remains prepared to react and respond to any emergency situation that may arise. This includes emergency preparedness training of facility personnel, facility support personnel, the training of, and coordination with, offsite emergency response organizations (EROs), and conducting periodic drills to ensure facility personnel and offsite response organization personnel training is maintained up to date.

N. Licensing Manager

The Licensing Manager reports to the HS&E Manager and has the responsibility for coordinating facility activities to ensure compliance is maintained with applicable Nuclear Regulatory Commission (NRC) requirements. The Licensing Manager is also responsible for ensuring abnormal events are reported to the NRC in accordance with NRC regulations.

O. Environmental Compliance Manager

The Environmental Compliance Manager reports to the HS&E Manager and has the responsibility for coordinating facility activities to ensure all local, state and federal environmental regulations are met. This includes submission of periodic reports to appropriate regulating organizations of effluents from the facility.

P. Radiation Protection Manager

The Radiation Protection Manager reports to the HS&E Manager and has the responsibility for implementing the Radiation Protection program. These duties include the training of personnel in use of equipment, control of radiation exposure of personnel, continuous determination of the radiological status of the facility, and conducting the radiological environmental monitoring program.

During emergency conditions the Radiation Protection Manager's duties may also include:

- Providing Emergency Operations Center personnel information and recommendations concerning chemical and radiation levels at the facility
- Gathering and compiling onsite and offsite radiological and chemical monitoring data
- Making recommendations concerning actions at the facility and offsite deemed necessary for limiting exposures to facility personnel and members of the general public
- Taking prime responsibility for decontamination activities.

In matters involving radiological protection, the Radiation Protection Manager has direct access to the Plant Manager.

Q. Industrial Safety Manager

The Industrial Safety Manager reports to the HS&E Manager and has the responsibility for the implementation of facility industrial safety programs and procedures. This shall include programs and procedures for training individuals in safety and maintaining the performance of the facility fire protection systems.

R. Criticality Safety Engineer

Criticality Safety Engineers report to the HS&E Manager (via a designated supervisory position, if applicable) and are responsible for the preparation and/or review of nuclear criticality safety evaluations and analyses, and conducting and reporting periodic nuclear criticality safety assessments. Nuclear criticality safety evaluations and analyses require independent reviews by a Criticality Safety Engineer.

S. Chemical Safety Engineer

The Chemical Safety Engineer reports to the HS&E Manager (via a designated supervisory position, if applicable) and is responsible for the preparation and/or review of chemical safety programs and procedures for the facility.

T. Shift Managers

The Shift Managers report to the Operations Manager and have the responsibility for ensuring safe operation of enrichment equipment and support equipment. Each Shift Manager directs assigned personnel in order to provide enrichment services in a safe, efficient manner.

U. Production Scheduling Manager

The Production Scheduling Manager reports to the Uranium Management Manager and has the responsibility for developing and maintaining production schedules for enrichment services.

V. Cylinder Management Manager

The Cylinder Management Manager reports to the Uranium Management Manager and has the responsibility for ensuring that cylinders of uranium hexafluoride are received and routed correctly at the facility, and is responsible for all transportation licensing.

W. Warehouse and Materials Manager

The Warehouse and Materials Manager reports to the Uranium Management Manager and has the responsibility for ensuring spare parts and other materials needed for operation of the facility are ordered, received, inspected and stored properly.

X. Safeguards Manager

The Safeguards Manager reports to the HS&E Manager and has the responsibility for ensuring the proper implementation of the FNMC Plan. This position is separate from and independent of the Operations, Technical Services, HS&E, and Human Resources departments to ensure a definite division between the safeguards group and the other departments. In matters involving safeguards, the Safeguards Manager has direct access to the Plant Manager.

Y. Chemistry Manager

The Chemistry Manager reports to the Technical Services Manager and has the responsibility for the implementation of chemistry analysis programs and procedures for the facility. This includes effluent sample collection, chemical analysis of effluents, comparison of effluent analysis results to limits, and reporting of chemical analysis of effluents to appropriate regulatory agencies.

Z. Performance Manager

The Performance Manager reports to the Technical Services Manager and has the responsibility for coordinating and maintaining testing programs for the facility. This includes testing of systems and components to ensure the systems and components are functioning as specified in design documents.

AA. Projects Manager

The Projects Manager reports to the Technical Services Manager and has the responsibility for the implementation of facility modifications and for maintaining the configuration management system. The Projects Manager also provides engineering support as needed to support facility operation and maintenance, and support of performance testing of systems and equipment.

BB. Engineering Manager

The Engineering Manager reports to the Technical Services Manager and has the responsibility for providing engineering support at the facility. This includes ensuring the safe operation of enrichment equipment and support equipment, providing maintenance support for equipment and systems, and developing operating and maintenance procedures for the facility. The Engineering Manager is responsible for the development of all design changes to the plant.

CC. Maintenance Manager

The Maintenance Manager reports to the Technical Services Manager and has the responsibility of directing and scheduling maintenance activities to ensure proper operation of the facility, including preparation and implementation of maintenance procedures. This includes activities such as repair and preventive maintenance of facility equipment. The Maintenance Manager also has the responsibility for coordinating and maintaining testing programs for the facility. This includes testing of systems and components to ensure the systems and components are functioning as specified in design documents.

DD. Administration Manager

The Administration Manager reports to the Human Resources Manager and has the responsibility for ensuring support functions such as accounting, word processing and general office management are provided for the facility.

EE. Community Relations Manager

The Community Relations Manager reports to the Human Resources Manager and has the responsibility for providing information about the facility and LES to the public and media. During an abnormal event at the facility, the Community Relations Manager ensures that the public and media receive accurate and up-to-date information.

FF. Security Manager

The Security Manager reports to the Human Resources Manager and has the responsibility for directing the activities of security personnel to ensure the physical protection of the facility. The Security Manager is also responsible for the protection of classified matter at the facility and obtaining security clearances for facility personnel and support personnel. In matters involving physical protection of the facility or classified matter, the Security Manager has direct access to the Plant Manager.

GG. Document Control Manager

The Document Control Manager reports to the Human Resources Manager and has the responsibility for adequately controlling documents at the facility.

HH. Training Manager

The Training Manager reports to the Human Resources Manager and has the responsibility for conducting training and maintaining training records for personnel at the facility.

2.2.2 Shift Crew Composition

The minimum operating shift crew consists of a Shift Manager (or Deputy Shift Manager in the absence of the Shift Manager), one Control Room operator, one Radiation Protection technician, one operator for each Cascade Hall and associated UF_6 handling systems, and security personnel. When only one Cascade Hall is in operation, a minimum of two operators is required.

At least one criticality safety engineer will be available, with appropriate ability to be contacted by the Shift Manager, to respond to any routine request or emergency condition. This availability may be offsite if adequate communication ability is provided to allow response as needed.

2.2.3 Safety Review Committee

The facility maintains a Safety Review Committee (SRC) to assist with the safe operation of the facility. The SRC shall report to the Plant Manager and shall provide technical and administrative review and audit of operations that could impact plant worker, public safety and environmental impacts. The scope of activities reviewed and audited by the SRC shall, as a minimum, include the following:

- Radiation protection
- Nuclear criticality safety
- Hazardous chemical safety
- Industrial safety including fire protection
- Environmental protection
- ALARA policy implementation
- Changes in facility design or operations.

The SRC shall conduct at least one facility audit per year for the above areas.

The Safety Review Committee shall be composed of at least five members, including the Chairman. Members of the SRC may be from the LES corporate office or technical staff. The five members shall include experts on operations and all safety disciplines (criticality, radiological, chemical, industrial). The Chairman, members and alternate members of the Safety Review Committee shall be formally appointed by the Plant Manager, shall have an academic degree in an engineering or physical science field; and, in addition, shall have a minimum of five years of technical experience, of which a minimum of three years shall relate directly to one or more of the safety disciplines (criticality, radiological, chemical, industrial).

The Safety Review Committee shall meet at least once per calendar quarter.

Review meetings shall be held within 30 days of any incident that is reportable to the NRC. These meetings may be combined with regular meetings. Following a reportable incident, the

SRC shall review the incident's causes, the responses, and both specific and generic corrective actions to ensure resolution of the problem is implemented.

A written report of each SRC meeting and audit shall be forwarded to the Plant Manager and appropriate Managers within 30 days and be retained in accordance with the records management system.

2.2.4 Personnel Qualification Requirements

The minimum qualification requirements for the facility functions that are directly responsible for its safe operation shall be as outlined below. These minimum qualifications were previously reviewed by the NRC staff and found to be acceptable (NRC, 1994).

The nuclear experience of each individual shall be determined to be acceptable by the Plant Manager. "Responsible nuclear experience" for these positions shall include (a) responsibility for and contributions towards support of facility(s) in the nuclear fuel cycle (e.g., design, construction, operation, and/or decommissioning), and (b) experience with chemical materials and/or processes. The Plant Manager may approve different experience requirements for key positions. Approval of different requirements shall be done in writing and only on a case-by-case basis.

The assignment of individuals to the Manager positions reporting directly to the Plant Manager, and to positions on the SRC, shall be approved by the Plant Manager. Assignments to all other staff positions shall be made within the normal administrative practices of the facility.

The actual qualifications of the individuals assigned to the key facility positions described in Section 2.2.1, Operating Organization will be maintained in the employee personnel files or other appropriate file at the facility. Development and maintenance of qualification records and training programs are the responsibility of the Human Resources Manager.

A. Chief Operating Officer

The President of LES, based on the individual's experience, proven ability in management of large-scale facilities, proven knowledge of regulatory and QA requirements, and overall leadership qualities, appoints the Chief Operating Officer.

B. Plant Manager

The Chief Operating Officer of LES shall appoint the Plant Manager as the overall manager of the facility. This appointment reflects confidence in the individual's ability as an effective programs and business manager. The Plant Manager shall be knowledgeable of the enrichment process, enrichment process controls and ancillary processes, criticality safety control, chemical safety, industrial safety, and radiation protection program concepts as they apply to the overall safety of a nuclear facility. The Plant Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and ten years of responsible nuclear experience.

C. Quality Assurance Director

The Quality Assurance Director shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and at least six years of responsible nuclear experience in the implementation of a quality assurance program. The QA Director shall have at least four years experience in a QA organization at a nuclear facility.

D. Quality Assurance Manager

The Quality Assurance (QA) Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and at least five years of responsible nuclear experience in the implementation of a quality assurance program. The QA Manager shall have at least two years experience in a QA organization at a nuclear facility.

E. Health, Safety, and Environment Manager

The Health, Safety, and Environment (HS&E) Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and at least five years of responsible nuclear experience in HS&E or related disciplines. The HS&E Manager shall also have at least one year of direct experience in the administration of nuclear criticality safety evaluations and analyses.

F. Operations Manager

The Operations Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience.

G. Uranium Management Manager

The Uranium Management Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience.

H. Technical Services Manager

The Technical Services Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience.

I. Human Resource Manager

The Human Resource Manager shall have as a minimum, a bachelor's degree in Personnel Management, Business Administration or related field, and three years of appropriate, responsible experience in implementing and supervising human resource responsibilities at an industrial facility.

J. Emergency Preparedness Manager

The Emergency Preparedness Manager shall have a minimum of five years of experience in the implementation and supervision of emergency plans and procedures at a nuclear facility. No credit for academic training may be taken toward fulfilling this experience requirement.

K. Licensing Manager

The Licensing Manager shall have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear licensing program.

L. Environmental Compliance Manager

The Environmental Compliance Manager shall have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear environmental compliance program.

M. Radiation Protection Manager

The Radiation Protection Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and three years of responsible nuclear experience associated with implementation of a Radiation Protection program. At least two years of experience shall be at a facility that processes uranium, including uranium in soluble form.

N. Industrial Safety Manager

The Industrial Safety Manager shall have, as a minimum, a bachelor's degree (or equivalent) in either an engineering or a scientific field and three years of appropriate, responsible nuclear experience associated with implementation of a facility safety program.

O. Criticality Safety Engineer

Criticality Safety Engineers shall have a minimum of two years experience in the implementation of a criticality safety program. These individuals shall hold a Bachelor of Science or Bachelor of Arts degree in an engineering or scientific field and have successfully completed a training program, applicable to the scope of operations, in the physics of criticality and in associated safety practices.

Should a change to the facility require a nuclear criticality safety evaluation or analysis, an individual who, as a minimum, possesses the equivalent qualifications of the Criticality Safety Engineer shall perform the evaluation or analysis. In addition, this individual shall have at least two years of experience performing criticality safety analyses and implementing criticality safety programs. An independent review of the evaluation or analysis, shall be performed by a qualified Criticality Safety Engineer.

P. Chemical Safety Engineer

The Chemical Safety Engineer shall have a minimum of two years experience in the preparation and/or review of chemical safety programs and procedures. This individual shall hold a bachelor's degree (or equivalent) in an engineering or scientific field and have successfully completed a training program, applicable to the scope of operations, in chemistry and in associated safety practices.

Q. Shift Managers

Shift Managers shall have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear operations program.

R. Production Scheduling Manager

The Production Scheduling Manager shall have a minimum of three years of appropriate, responsible experience in implementing and supervising a continuous production scheduling program.

S. Cylinder Management Manager

The Cylinder Management Manager shall have a minimum of three years of appropriate, responsible experience in implementing and supervising a continuous production scheduling program.

T. Warehouse and Materials Manager

The Warehouse and Materials Manager shall have a minimum of three years of appropriate, responsible experience in implementing and supervising a purchasing and inventory program.

U. Safeguards Manager

The Safeguards Manager shall have as a minimum, a bachelor's degree in an engineering or scientific field, and five years of experience in the management of a safeguards program for Special Nuclear Material, including responsibilities for material control and accounting. No credit for academic training may be taken toward fulfilling this experience requirement.

V. Chemistry Manager

The Chemistry Manager shall have, as a minimum, a bachelor's degree (or equivalent) in either an engineering or a scientific field and three years of appropriate, responsible nuclear experience associated with implementation of a facility chemistry program.

W. Projects Manager

The Projects Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and have a minimum of five years of appropriate, responsible nuclear experience.

X. Engineering Manager

The Engineering Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear engineering program.

Y. Maintenance Manager

The Maintenance Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience.

Z. Administration Manager

The Administration Manager shall have a minimum of three years of appropriate, responsible experience in implementing and supervising administrative responsibilities at an industrial facility.

AA. Community Relations Manager

The Community Relations Manager shall have as a minimum, a bachelor's degree in Public Relations, Political Science or Business Administration and three years of appropriate, responsible experience in implementing and supervising a community relations program.

BB. Security Manager

The Security Manager shall have as a minimum, a bachelor's degree in an engineering or scientific field, and five years of experience in the responsible management of physical security at a facility requiring security capability similar to that required for the facility. No credit for academic training may be taken toward fulfilling this experience requirement.

CC. Document Control Manager

The Document Control Manager shall have a minimum of three years of appropriate, responsible experience in implementing and supervising a document control program.

DD. Training Manager

The Training Manager shall have a minimum of five years of appropriate, responsible experience in implementing and supervising a training program.

EE. Performance Manager

The Performance Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience.

2.3 ADMINISTRATION

This section summarizes how the activities that are essential for implementation of the management measures and other HS&E functions are documented in formally approved, written procedures, prepared in compliance with a formal document control program. The mechanism for reporting potentially unsafe conditions or activities to the HS&E organization and facility management is also summarized.

The management measures summarized below are the same management measures LES submitted in the license application for the Claiborne Enrichment Center (LES, 1993). The NRC staff documented their review and acceptance of these management measures in NUREG-1491 (NRC, 1994). Details of the management measures are provided in Chapter 11, Management Measures.

2.3.1 Configuration Management

Configuration management is provided for Items Relied On For Safety (IROFS) throughout facility design, construction, testing, and operation. Configuration management provides the means to establish and maintain a technical baseline for the facility based on clearly defined requirements. During design and construction, the Engineering and Contracts Manager has responsibility for configuration management through the design control process. Selected documentation is controlled under the configuration management system in accordance with appropriate QA procedures associated with design control, document control, and records management. Design changes to IROFS undergo formal review, including interdisciplinary reviews as appropriate, in accordance with these procedures.

Configuration management provides the means to establish and maintain the essential features of the design basis of IROFS. As the project progresses from design and construction to operation, configuration management is maintained by the facility engineering organization as the overall focus of activities changes.

Additional details on Configuration Management are provided in Chapter 11, Management Measures.

2.3.2 Maintenance

The maintenance program will be implemented for the operations phase of the facility. Preventive maintenance activities, surveillance, and performance trending provide reasonable and continuing assurance that IROFS will be available and reliable to perform their safety functions.

The purpose of planned and scheduled maintenance for IROFS is to ensure that the equipment and controls are kept in a condition of readiness to perform the planned and designed functions when required. Appropriate plant management is responsible for ensuring the operational readiness of IROFS under this control. For this reason, the maintenance function is administratively closely coupled to operations. The maintenance organization plans, schedules, tracks, and maintains records for maintenance activities.

Maintenance activities generally fall into the following categories:

- Corrective maintenance
- Preventive maintenance
- Surveillance/monitoring
- Functional testing.

These maintenance categories are discussed in detail in Chapter 11, Management Measures.

2.3.3 Training and Qualifications

Formal planned training programs shall be established for facility employees. Indoctrination training shall be provided to employees within 30 days of reporting to work, and shall address safety preparedness for all safety disciplines (criticality, radiological, chemical, industrial), ALARA practices, and emergency procedures. In-depth training programs shall be provided to individuals depending on job requirements in the areas of radiological safety (for all personnel with access to the Restricted Area) and in criticality safety control. Nuclear criticality safety training shall satisfy the recommendations of ANSI/ANS-8.20 - 1991, Nuclear Criticality Safety Training (ANSI, 1991). Retraining of personnel previously trained shall be performed for radiological and criticality safety at least annually, and shall include updating and changes in required skills. The training program shall include methods for verifying training effectiveness, such as written tests, actual demonstration of skills, and where required by regulation, maintaining a current and valid license demonstrating qualification. Changes to training shall be implemented if indicated due to incidents potentially compromising safety, or if changes are made to facilities or processes.

The training programs and maintenance of the training program records at the facility are the responsibility of the Human Resources Manager. Accurate records are maintained on each employee's qualifications, experience, training and retraining. The employee training file shall include records of all general employee training, technical training, and employee development training conducted at the facility. The employee training file shall also contain records of special company sponsored training conducted by others. The training records for each individual are maintained so that they are accurate and retrievable. Training records are retained in accordance with the records management system.

Additional details on the facility training program are provided in Chapter 11, Management Measures.

2.3.4 Procedures

Activities involving licensed materials will be conducted through the use of approved, written procedures. Applicable procedure and training requirements will be satisfied before use of the procedure. Procedures will be used to control activities in order to ensure the activities are carried out in a safe manner.

Generally, four types of plant procedures are used to control activities: operating procedures, administrative procedures, maintenance procedures, and emergency procedures. Operating procedures, developed for workstation and control room operators, are used to directly control process operations. Administrative procedures are written by each department as necessary to control activities that support process operations, including management measures (e.g. configuration management, training and record-keeping). Maintenance procedures address preventive and corrective maintenance, surveillance (includes calibration, inspection, and other surveillance testing), functional testing following maintenance, and requirements for pre-maintenance activity involving reviews of the work to be performed and reviews of procedures. Emergency procedures address the preplanned actions of operators and other plant personnel in the event of an emergency.

Policies and procedures will be developed to ensure that there are ties between major plant safety functions such as the ISA, management measures for items relied on for safety (IROFS), radiation safety, nuclear criticality safety, fire safety, chemical safety, environmental monitoring, and emergency planning.

Chapter 11 details the use of procedures, including development, revision, and distribution and control.

2.3.5 Audits and Assessments

The LES QA Program requires periodic audits to confirm that activities affecting quality comply with the QA Program and that the QA Program is being implemented effectively. The assessment function includes audits and other independent assessments to verify performance. These assessments provide a comprehensive independent evaluation of activities, including activities delegated to others under the LES QA Program, and procedures. Personnel who do not have direct responsibility in the area being assessed conduct these assessments.

An assessment and audit program for operational quality assurance of the enrichment facility is established, and periodically reviewed by management, to:

- verify that the configuration and operation of the facility are consistent with LES company policy, approved procedures and license provisions
- review important proposed facility modifications, tests and procedures
- verify that reportable occurrences are investigated and corrected in a manner which reduces the probability of recurrence of such events
- to detect trends which may not be apparent to a day-to-day observer.

The organizational structure for conducting the operational reviews and audit program includes:

- The Safety Review Committee appointed by the Plant Manager
- Regular audits conducted by the Quality Assurance Department.

Each of the above shall have the authority necessary to discharge its responsibilities adequately. Implicit in this authority shall be access to facility records and personnel as required in order to perform reviews and audits properly.

Additional details on audits and assessments are provided in Chapter 11, Management Measures.

2.3.5.1 Safety Review Committee

The Safety Review Committee (SRC) provides technical and administrative review of facility operations that could impact plant worker and public safety. Details on the SRC and the scope of activities reviewed by the SRC are provided in Section 2.2.3, Safety Review Committee.

2.3.5.2 Quality Assurance Department

The Quality Assurance Department conducts periodic audits of activities associated with the facility, in order to verify the facility's compliance with established procedures. The LES Quality Assurance Program Description is included in Chapter 11,Management Measures as Appendix A.

2.3.5.3 Facility Operating Organization

The facility operating organization shall provide, as part of the normal duties of supervisory personnel, timely and continuing monitoring of operating activities to assist the Plant Manager in keeping abreast of general facility conditions and to verify that the day-to-day operating activities are conducted safely and in accordance with applicable administrative controls.

These continuing monitoring activities are considered to be an integral part of the routine supervisory function and are important to the safety of the facility operation.

2.3.5.4 Audited Organizations

Audited organizations shall assure that deficiencies identified are corrected in a timely manner.

Audited organizations shall transmit a response to each audit report within the time period specified in the audit. For each identified deficiency, the response shall identify the corrective action taken or to be taken. For each identified deficiency, the response shall also address whether or not the deficiency is considered to be indicative of other problems (e.g., a specific audit finding may indicate a generic problem) and the corrective action taken or to be taken for any such problems determined.

Copies of audit reports and responses are maintained in accordance with the records management system.

2.3.6 Incident Investigations

Abnormal events that potentially threaten or lessen the effectiveness of health, safety or environmental protection are identified and reported to the HS&E Manager or designee through the Corrective Action Program (CAP) which is described in more detail in Chapter 11, Management Measures. Each event is considered in terms of its requirements for reporting in accordance with regulations and is evaluated to determine the level of investigation required. These evaluations and investigations are conducted in accordance with approved CAP procedures. The depth of the investigation depends upon the severity of the incident in terms of the levels of uranium released and/or the degree of potential for exposure of workers, the public or the environment.

The HS&E Manager, or designee is responsible for:

- maintaining a list of agencies to be notified
- determining if a report to an agency is required
- notifying the agency when required.

The licensing function has the responsibility for continuing communications with government agencies and tracking corrective actions to completion.

The process of incident identification, investigation, root cause analysis, environmental protection analysis, recording, reporting, and follow-up shall be addressed in and performed in accordance with written procedures. Radiological, criticality, hazardous chemical, and industrial safety requirements shall be addressed. Guidance for classifying incidents shall be contained in facility procedures, including a list of threshold off-normal incidents.

The HS&E Manager or designee shall, through implementation of the CAP, maintain a record of corrective actions to be implemented as a result of off-normal investigations. These corrective actions shall include documenting lessons learned, and implementing worker training where indicated, and shall be tracked to completion by the HS&E Manager or designee within the CAP.

Additional details on incident investigations are provided in Chapter 11, Management Measures.

2.3.7 Employee Concerns

Employees who feel that safety or quality is being compromised have the right and responsibility to initiate the "stop work" process in accordance with the applicable project or facility procedures to ensure the work environment is placed in a safe condition.

Employees also have access to various resources to ensure their safety or quality concerns are addressed, including:

- line management or other facility management (e.g., HS&E Manager, Plant Manager, QA Manager)
- the facility safety organization (i.e., any of the safety engineers or managers)
- NRC's requirements under 10 CFR 19, Notices, Instructions and Reports to Workers: Inspection and Investigations (CFR, 2003a)
- LES CAP - a simple mechanism available for use by any person at the NEF site for reporting unusual events and potentially unsafe conditions or activities.

2.3.8 Records Management

Procedures are established which control the preparation and issuance of documents such as manuals, instructions, drawings, procedures, specifications, and supplier-supplied documents, including any changes thereto. Measures are established to ensure documents, including revisions, are adequately reviewed, approved, and released for use by authorized personnel.

Document control procedures require documents to be transmitted and received in a timely manner at appropriate locations including the location where the prescribed activity is to be performed. Controlled copies of these documents and their revisions are distributed to and used by the persons performing the activity.

Superseded documents are destroyed or are retained only when they have been properly labeled. Indexes of current documents are maintained and controlled.

The QA Program assigns responsibility for verifying QA record retention to the QA Manager. Applicable design specifications, procurement documents, or other documents specify the QA records to be generated by, supplied to, or held, in accordance with approved procedures. QA records are not considered valid until they are authenticated and dated by authorized personnel.

Additional details on the records management program are provided in Chapter 11, Management Measures.

2.3.9 Written Agreements with Offsite Emergency Resources

The plans for coping with emergencies at the facility are presented in detail in the Emergency Plan. The Emergency Plan includes a description of the facility emergency response organization and interfaces with off-site EROs. Written agreements between the facility and off-site EROs, including the local fire department, the local law enforcement agency, ambulance/rescue units, and medical services and facilities have been established.

Coordination with participating government agencies (State, Counties) is vital to the safety and health of plant personnel and the general public. The principal state and local agencies/organizations having responsibilities for radiological or other hazardous material emergencies for the facility are:

A. New Mexico Department of Public Safety, Office of Emergency Management

B. Eunice Emergency Response Services

C. Hobbs Emergency Response Services

Details of the interfaces with these agencies are provided in Section 4 of the Emergency Plan.

2.4 REFERENCES

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CFR, 2003b. Title 10, Code of Federal Regulations, Part 40, Domestic Licensing of Source Material, 2003.

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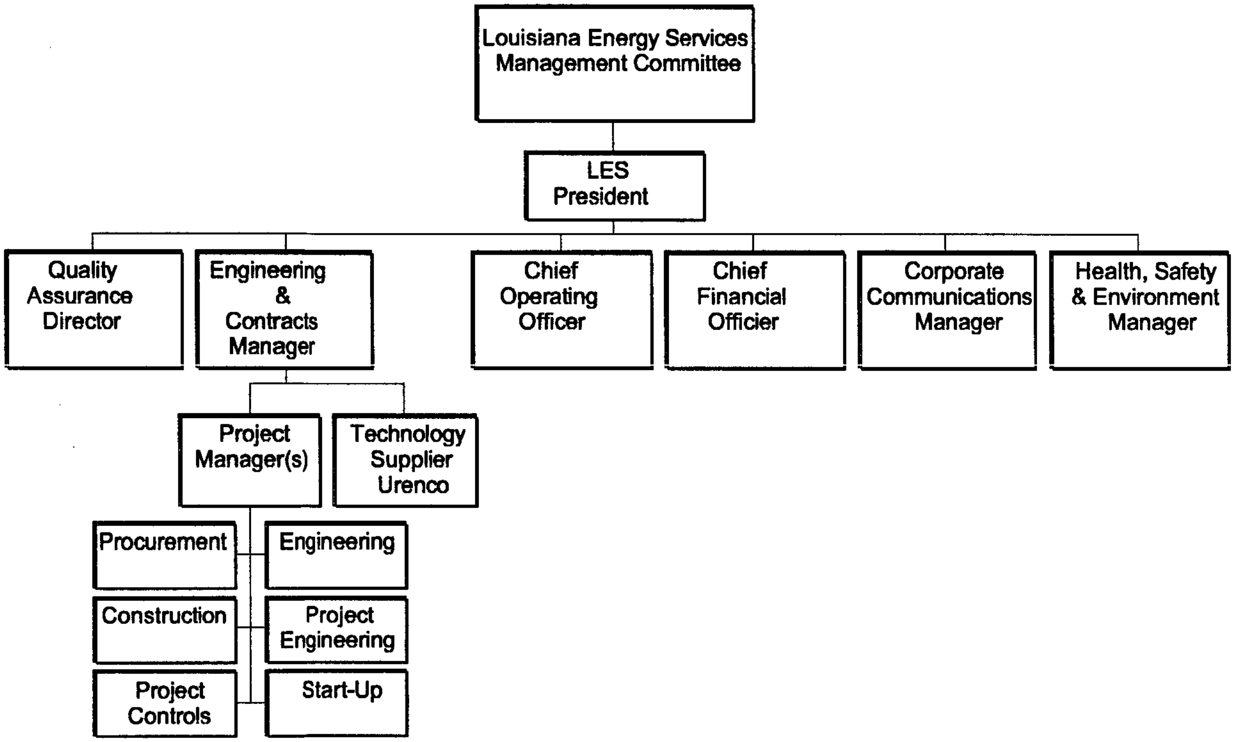
LES, 1993. Claiborne Enrichment Center Safety Analysis Report, Chapter 11, Louisiana Energy Services, December 1993.

NRC, 1992. Proposed Method for Regulating Major Materials Licensees, NUREG-1324, U.S. Nuclear Regulatory Commission, 1992.

NRC, 1994. Safety Evaluation Report for the Claiborne Enrichment Center, Homer, Louisiana, NUREG-1491, Section 10, U.S. Nuclear Regulatory Commission, January 1994.

NRC, 2002. Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, NUREG-1520, U.S. Nuclear Regulatory Commission, March 2002.

FIGURES



4

REFERENCE NUMBER
Figure 2.1–1.dwg

FIGURE 2.1-1
LES CORPORATE, DESIGN AND CONSTRUCTION ORGANIZATION

REVISION 4 APRIL 2005

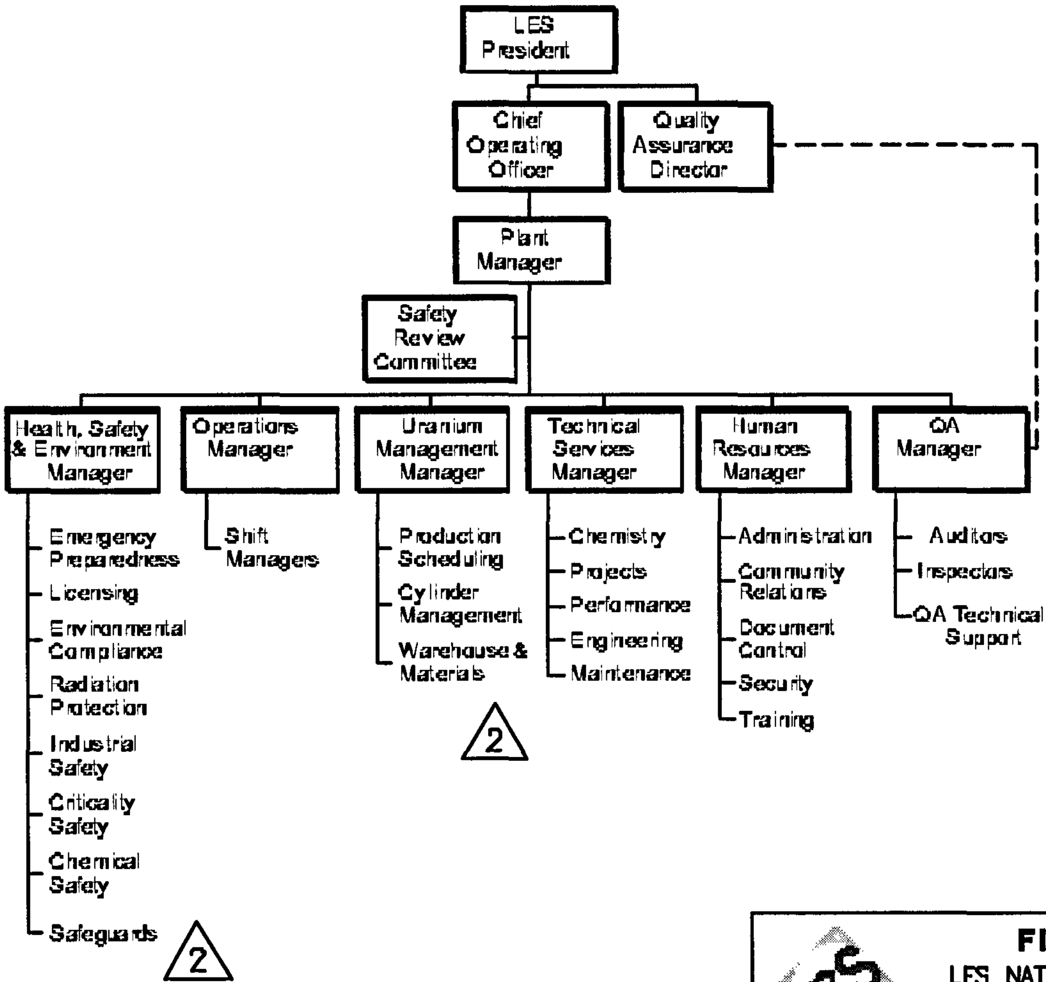


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Figure 2.1-2.dwg

FIGURE 2.1-2

LES NATIONAL ENRICHMENT FACILITY
OPERATING ORGANIZATION

REVISION 2 DATE: JULY 2004

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3.0 SAFETY PROGRAM COMMITMENTS

This section presents the commitments pertaining to the facility's safety program including the performance of an ISA. 10 CFR Part 70 (CFR, 2003b) contains a number of specific safety program requirements related to the integrated safety analysis (ISA). These include the primary requirements that an ISA be conducted, and that it evaluate and show that the facility complies with the performance requirements of 10 CFR 70.61 (CFR, 2003c).

3.0 SAFETY PROGRAM

The three elements of the safety program defined in 10 CFR 70.62(a) (CFR, 2003d) are addressed below.

3.0.1 Process Safety Information

A. LES has compiled and maintains up-to-date documentation of process safety information. Written process-safety information is used in updating the ISA and in identifying and understanding the hazards associated with the processes. The compilation of written process-safety information includes information pertaining to:

1. The hazards of all materials used or produced in the process, which includes information on chemical and physical properties such as are included on Material Safety Data Sheets meeting the requirements of 29 CFR 1910.1200(g) (CFR, 2003e).

2. Technology of the process which includes block flow diagrams or simplified process flow diagrams, a brief outline of the process chemistry, safe upper and lower limits for controlled parameters (e.g., temperature, pressure, flow, and concentration), and evaluation of the health and safety consequences of process deviations.

3. Equipment used in the process including general information on topics such as the materials of construction, piping and instrumentation diagrams (P&IDs), ventilation, design codes and standards employed, material and energy balances, IROFS (e.g., interlocks, detection, or suppression systems), electrical classification, and relief system design and design basis.

The process-safety information described above is maintained up-to-date by the configuration management program described in Section 11.1, Configuration Management.

B. LES has developed procedures and criteria for changing the ISA. This includes implementation of a facility change mechanism that meets the requirements of 10 CFR 70.72 (CFR, 2003f).

The development and implementation of procedures is described in Section 11.4, Procedures Development and Implementation.

C. LES uses personnel with the appropriate experience and expertise in engineering and process operations to maintain the ISA. The ISA Team for the various processes consists of individuals who are knowledgeable in the ISA method(s) and the operation, hazards, and safety design criteria of the particular process. Training and qualifications of individuals responsible for maintaining the ISA are described in Section 11.3, Training and Qualifications, Section 2.2, Key Management Positions, and Section 3.2, Integrated Safety Analysis Team.

3.0.2 Integrated Safety Analysis

A. LES has conducted an ISA for each process, such that it identifies (i) radiological hazards, (ii) chemical hazards that could increase radiological risk, (iii) facility hazards that could increase radiological risk, (iv) potential accident sequences, (v) consequences and likelihood of each accident sequence and (vi) IROFS including the assumptions and conditions under which they support compliance with the performance requirements of 10 CFR 70.61 (CFR, 2003c).

A synopsis of the results of the ISA, including the information specified in 10 CFR 70.65(b) (CFR, 2003a), is provided in the National Enrichment Facility Integrated Safety Analysis Summary.

B. LES has implemented programs to maintain the ISA and supporting documentation so that it is accurate and up-to-date. Changes to the ISA Summary are submitted to the NRC, in accordance with 10 CFR 70.72(d)(1) and (3) (CFR, 2003f). The ISA update process accounts for any changes made to the facility or its processes. This update will also verify that initiating event frequencies and IROFS reliability values assumed in the ISA remain valid. Any changes required to the ISA as a result of the update process will be included in a revision to the ISA. Management policies, organizational responsibilities, revision time frame, and procedures to perform and approve revisions to the ISA are outlined in Chapter 11.0, Management Measures. Evaluation of any facility changes or changes in the process safety information that may alter the parameters of an accident sequence is by the ISA method(s) as described in the ISA Summary Document. For any revisions to the ISA, personnel having qualifications similar to those of ISA team members who conducted the original ISA are used.

C. Personnel used to update and maintain the ISA and ISA Summary are trained in the ISA method(s) and are suitably qualified. Training and Qualification of personnel used to update or maintain the ISA are described in Section 11.3, Training and Qualifications.

D. Proposed changes to the facility or its operations are evaluated using the ISA method(s). New or additional IROFS and appropriate management measures are designated as required. The adequacy of existing IROFS and associated management measures are promptly evaluated to determine if they are impacted by changes to the facility and/or its processes. If a proposed change results in a new type of accident sequence or increases the consequences or likelihood of a previously analyzed accident sequence within the context of 10 CFR 70.61 (CFR, 2003c), the adequacy of existing IROFS and associated management measures are promptly evaluated and the necessary changes are made, if required.

E. Unacceptable performance deficiencies associated with IROFS are addressed that are identified through updates to the ISA.

F. Written procedures are maintained on site. Section 11.4, Procedures Development and Implementation, discusses the procedures program.

G. All IROFS are maintained so that they are available and reliable when needed.

3.0.3 Management Measures

Management measures are functions applied to IROFS, and any items that may affect the function of IROFS. IROFS management measures ensure compliance with the performance requirements assumed in the ISA documentation. The measures are applied to particular structures, systems, equipment, components, and activities of personnel, and may be graded commensurate with the reduction of the risk attributable to that IROFS. The IROFS management measures shall ensure that these structures, systems, equipment, components, and activities of personnel within the identified IROFS boundary are designed, implemented, and maintained, as necessary, to ensure they are available and reliable to perform their function when needed, to comply with the performance requirements assumed in the ISA documentation.

The following types of management measures are required by the 10 CFR 70.4 (CFR, 2003b) definition of management measures. The description for each management measure reflects the general requirements applicable to each IROFS. Any management measure that deviates from the general requirements described in this section, which are consistent with the performance requirements assumed in the ISA documentation, are discussed in the National Enrichment Facility Integrated Safety Analysis Summary.

Configuration Management

The configuration management program is required by 10 CFR 70.72 (CFR, 2003f) and establishes a system to evaluate, implement, and track each change to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel. Configuration management of IROFS, and any items that may affect the function of IROFS, is applied to all items identified within the scope of the IROFS boundary. Any change to structures, systems, equipment, components, and activities of personnel within the identified IROFS boundary must be evaluated before the change is implemented. If the change requires an amendment to the License, Nuclear Regulatory Commission approval is required prior to implementation.

Maintenance

Maintenance of IROFS, and any items that may affect the function of IROFS, encompasses planned surveillance testing and preventative maintenance, as well as unplanned corrective maintenance. Implementation of approved configuration management changes to hardware is also generally performed as a planned maintenance function.

Planned surveillance testing (e.g., functional/performance testing, instrument calibrations) monitors the integrity and capability of IROFS, and any items that may affect the function of IROFS, to ensure they are available and reliable to perform their function when needed, to comply with the performance requirements assumed in the ISA documentation. All necessary periodic surveillance testing is generally performed on an annual frequency (any exceptions credited within the ISA are discussed in the National Enrichment Facility Integrated Safety Analysis Summary).

Planned preventative maintenance (PM) includes periodic refurbishment, partial or complete overhaul, or replacement of IROFS, as necessary, to ensure the continued availability and reliability of the safety function assumed in the ISA documentation. In determining the frequency of any PM, consideration is given to appropriately balancing the objective of preventing failures through maintenance, against the objective of minimizing unavailability of IROFS because of PM. In addition, feedback from PM and corrective maintenance and the results of incident investigations and identified root causes are used, as appropriate, to modify the frequency or scope of PM.

Planned maintenance on IROFS, or any items that may affect the function of IROFS, that do not have redundant functions available, will provide for compensatory measures to be put into place to ensure that the IROFS function is performed until it is put back into service.

Corrective maintenance involves repair or replacement of equipment that has unexpectedly degraded or failed. Corrective maintenance restores the equipment to acceptable performance through a planned, systematic, controlled, and documented approach for the repair and replacement activities.

Following any maintenance on IROFS, and before returning an IROFS to operational status, functional testing of the IROFS, as necessary, is performed to ensure the IROFS is capable of performing its intended safety function.

Training and Qualifications

IROFS, and any items that may affect the function of IROFS, require that personnel involved at each level (from design through and including any assumed process implementation steps or actions) have and maintain the appropriate training and qualifications. Employees are provided with formal training to establish the knowledge foundation and on-the-job training to develop work performance skills. For process implemented steps or actions, a needs/job analysis is performed and tasks are identified to ensure that appropriate training is provided to personnel working on tasks related to IROFS. Minimum training requirements are developed for those positions whose activities are relied on for safety. Initial identification of job-specific training requirements is based on experience. Entry-level criteria (e.g., education, technical background, and/or experience) for these positions are contained in position descriptions.

Qualification is indicated by successful completion of prescribed training, demonstration of the ability to perform assigned tasks, and where required by regulation, maintaining a current and valid license or certification.

Continuing training is provided, as required, to maintain proficiency in specific knowledge and skill related activities. For all IROFS, and any items that may affect the function of IROFS, involving process implemented steps or actions, annual refresher training or requalification is generally required (any exceptions credited within the ISA are discussed in the National Enrichment Facility Integrated Safety Analysis Summary).

Procedures

All activities involving IROFS, and any items that may affect the function of IROFS, are conducted in accordance with approved procedures. Each of the other IROFS management measures (e.g., configuration management, maintenance, training) is implemented via approved procedures. These procedures are intended to provide a pre-planned method of conducting the activity in order to eliminate errors due to on-the-spot analysis and judgments.

All procedures are sufficiently detailed that qualified individuals can perform the required functions without direct supervision. However, written procedures cannot address all contingencies and operating conditions. Therefore, they contain a degree of flexibility appropriate to the activities being performed. Procedural guidance exists to identify the manner in which procedures are to be implemented. For example, routine procedural actions may not require the procedure to be present during implementation of the actions, while complex jobs, or checking with numerous sequences may require valve alignment checks, approved operator aids, or in-hand procedures that are referenced directly when the job is conducted.

To support the requirement to minimize challenges to IROFS, and any items that may affect the function of IROFS, specific procedures for abnormal events are also provided. These procedures are based on a sequence of observations and actions to prevent or mitigate the consequences of an abnormal situation.

Audits and Assessments

Audits are focused on verifying compliance with regulatory and procedural requirements and licensing commitments. Assessments are focused on effectiveness of activities and ensuring that IROFS are reliable and are available to perform their intended safety functions as documented in the ISA. The frequency of audits and assessments is based upon the status and safety importance of the activities being performed and upon work history. However, at a minimum, all activities associated with maintaining IROFS will generally be audited or assessed on an annual basis (any exceptions credited within the ISA are discussed in the National Enrichment Facility Integrated Safety Analysis Summary).

Incident Investigations

Incident investigations are conducted within the Corrective Action Program (CAP). Incidents associated with IROFS, and any items that may affect the function of IROFS, encompass a range of items, including (a) processes that behave in unexpected ways, (b) procedural activities not performed in accordance with the approved procedure, (c) discovered deficiency, degradation, or non-conformance with an IROFS, or any items that may affect the function of IROFS. Additionally, audit and assessment results are tracked in the Corrective Action Program.

Feedback from the results of incident investigations and identified root causes are used, as appropriate, to modify management measures to provided continued assurance that the reliability and availability of IROFS remain consistent with the performance requirements assumed in the ISA documentation.

Records Management

All records associated with IROFS, and any items that may affect the function of IROFS, shall be managed in a controlled and systematic manner in order to provide identifiable and retrievable documentation. Applicable design specifications, procurement documents, or other documents specify the QA records to be generated by, supplied to, or held, in accordance with approved procedures are included.

Other Quality Assurance Elements

Other quality assurance elements associated with IROFS, or any items that may affect the function of IROFS, that are required to ensure the IROFS is available and reliable to perform the function when needed to comply with the performance requirements assumed in the ISA documentation, are discussed in the National Enrichment Facility Integrated Safety Analysis Summary.

3.1 INTEGRATED SAFETY ANALYSIS METHODS

This section outlines the approach utilized for performing the integrated safety analysis (ISA) of the process accident sequences. The approach used for performing the ISA is consistent with Example Procedure for Accident Sequence Evaluation, Appendix A to Chapter 3 of NUREG-1520 (NRC, 2002a). This approach employs a semi-quantitative risk index method for categorizing accident sequences in terms of their likelihood of occurrence and their consequences of concern. The risk index method framework identifies which accident sequences have consequences that could exceed the performance requirements of 10 CFR 70.61 (CFR, 2003c) and, therefore, require designation of items relied on for safety (IROFS) and supporting management measures. Descriptions of these general types of higher consequence accident sequences are reported in the ISA Summary.

The ISA is a systematic analysis to identify plant and external hazards and the potential for initiating accident sequences, the potential accident sequences, the likelihood and consequences, and the IROFS.

The ISA uses a hazard analysis method to identify the hazards which are relevant for each system or facility. The ISA Team reviewed the hazard identified for the "credible worst-case" consequences. All credible high or intermediate severity consequence accident scenarios were assigned accident sequence identifiers, accident sequence descriptions, and a risk index determination was made.

The risk index method is regarded as a screening method, not as a definitive method of proving the adequacy or inadequacy of the IROFS for any particular accident.

The tabular accident summary resulting from the ISA identifies, for each sequence, which engineered or administrative IROFS must fail to allow the occurrence of consequences that exceed the levels identified in 10 CFR 70.61 (CFR, 2003c).

For this license application, two ISA Teams were formed. This was necessary because the sensitive nature of some of the facility design information related to the enrichment process required the use of personnel with the appropriate national security clearances. This team performed the ISA on the Cascade System, Contingency Dump System, Centrifuge Test System and the Centrifuge Post Mortem System. This ISA Team is referred to as the Classified ISA Team. The Non-Classified Team, referred to in the remainder of this text as the ISA Team, performed the ISA on the remainder of the facility systems and structures. In addition, the (non-classified) ISA Team performed the External Events and Fire Hazard Assessment for the entire facility.

In preparing for the ISA, the Accident Analysis in the Safety Analysis Report (LES, 1993) for the Claiborne Enrichment Center was reviewed. In addition, experienced personnel with familiarity with the gas centrifuge enrichment technology safety analysis where used on the ISA Team. This provides a good peer check of the final ISA results.

A procedure was developed to guide the conduct of the ISA. This procedure was used by both teams. In addition, there were common participants on both teams to further integrate the approaches employed by both teams. These steps were taken to ensure the consistency of the results of the two teams. A non-classified summary of the results of the Classified ISA has been prepared and incorporated into the ISA Summary.

3.1.1 Hazard Identification

The hazard and operability (HAZOP) analysis method was used for identifying the hazards for the Uranium Hexafluoride (UF_6) process systems and Technical Services Building systems. This method is consistent with the guidance provided in NUREG-1513 (NRC, 2001a) and NUREG-1520 (NRC, 2002a). The hazards identification process results in identification of physical, radiological or chemical characteristics that have the potential for causing harm to site workers, the public, or to the environment. Hazards are identified through a systematic review process that entails the use of system descriptions, piping and instrumentation diagrams, process flow diagrams, plot plans, topographic maps, utility system drawings, and specifications of major process equipment. In addition, criticality hazards identification were performed for the areas of the facility where fissile material is expected to be present. The criticality safety analyses contain information about the location and geometry of the fissile material and other materials in the process, for both normal and credible abnormal conditions. The ISA input information is included in the ISA documentation and is available to be verified as part of an on-site review.

The hazard identification process documents materials that are:

- Radioactive
- Fissile
- Flammable
- Explosive
- Toxic
- Reactive.

The hazard identification also identifies potentially hazardous process conditions. Most hazards were assessed individually for the potential impact on the discrete components of the process systems. However, for hazards from fires (external to the process system) and external events (seismic, severe weather, etc.), the hazards were assessed on a facility wide basis.

For the purpose of evaluating the impacts of fire hazards, the ISA team considered the following:

- Postulated the development of a fire occurring in in-situ combustibles from an unidentified ignition source (e.g., electrical shorting, or other source)
- Postulated the development of a fire occurring in transient combustibles from an unidentified ignition source (e.g., electrical shorting, or other source)
- Evaluated the uranic content in the space and its configuration (e.g., UF_6 solid/gas in cylinders, UF_6 gas in piping, UF_6 and/or byproducts bound on chemical traps, Uranyl Fluoride (UO_2F_2) particulate on solid waste or in solution). The appropriate configuration was considered relative to the likelihood of the target releasing its uranic content as a result of a fire in the area.

In order to assess the potential severity of a given fire and the resulting failures to critical systems, the facility Fire Hazard Analysis was consulted. However, since the design supporting the license submittal for this facility is not yet at the detailed design stage, detailed in-situ combustible loading and in-situ combustible configuration information is not yet available. Therefore, in order to place reasonable and conservative bounds on the fire scenarios analyzed, the ISA Team estimated in-situ combustible loadings based on information of the in-situ combustible loading from Urenco's Almelo SP-5 plant (on which the National Enrichment Facility (NEF) design is based). This information from SP-5 indicates that in-situ combustible loads are expected to be very low.

The Fire Safety Management Program will limit the allowable quantity of transient combustibles in critical plant areas (i.e., uranium areas). Nevertheless, the ISA Team still assumed the presence of moderate quantities of ordinary (Class A) combustibles (e.g., trash, packing materials, maintenance items or packaging, etc.) in excess of anticipated procedural limits. This was not considered a failure of the associated administrative IROFS feature for controlling/ minimizing transient combustible loading in all radiation/uranium areas. Failure of the IROFS is connoted as the presence of extreme or severe quantities of transients (e.g., large piles of combustible solids, bulk quantities of flammable/combustible liquids or gases, etc.). The Urenco ISA Team representatives all indicated that these types of transient combustible conditions do not occur in the European plants. Accordingly, and given the orientation and training that facility employees will receive indicating that these types of fire hazards are unacceptable, the administrative IROFS preventing severe accumulations has been assigned a high degree of reliability.

Fires that involve additional in-situ or transient combustibles from outside each respective fire area could result in exposure of additional uranic content being released in a fire beyond the quantities assumed above. For this reason, fire barriers are needed to ensure that fires cannot propagate from non-uranium containing areas into uranium (U) areas or from one U area to another U area (unless the uranium content in the space is insignificant, i.e., would be a low consequence event). Fire barriers shall be designed with adequate safety margin such that the total combustible loading (in-situ and transient) allowed to expose the barrier will not exceed 80% of the hourly fire resistance rating of the barrier.

For external events, the impacts were evaluated for the following hazards:

External events were considered at the site and facility level versus at individual system nodes. Specific external event HAZOP guidewords were developed for use during the external event portion of the ISA. The external event ISA considered both natural phenomena and man-made hazards. During the external event ISA team meeting, each area of the plant was discussed as to whether or not it could be adversely affected by the specific external event under consideration. If so, specific consequences were then discussed. If the consequences were known or assumed to be high, then a specific design basis with a likelihood of highly unlikely would be selected.

Given that external events were considered at the facility level, the ISA for external events was performed after the ISA team meetings for all plant systems were completed. This provided the best opportunity to perform the ISA at the site or facility level. Each external event was assessed for both the uncontrolled case and then for the controlled case. The controlled cases could be a specific design basis for that external event, IROFS or a combination of both. An Accident Sequence and Risk matrix was prepared for each external event.

External events evaluated included:

- Seismic

- Tornado, Tornado Missile and High Wind
- Snow and Ice
- Flooding
- Local Precipitation
- Other (Transportation and Nearby Facility Accidents)
- Aircraft
- Pipelines
- Highway
- Other Nearby Facilities
- Railroad
- On-site Use of Natural Gas
- Internal Flooding from On-Site Above Ground Liquid Storage Tanks.

The ISA is intended to give assurance that the potential failures, hazards, accident sequences, scenarios, and IROFS have been investigated in an integrated fashion, so as to adequately consider common mode and common cause situations. Included in this integrated review is the identification of IROFS function that may be simultaneously beneficial and harmful with respect to different hazards, and interactions that might not have been considered in the previously completed sub-analyses. This review is intended to ensure that the designation of one IROFS does not negate the preventive or mitigation function of another IROFS. An integration checklist is used by the ISA Team as a guide to facilitate the integrated review process.

Some items that warrant special consideration during the integration process are:

- Common mode failures and common cause situations.
- Support system failures such as loss of electrical power or city water. Such failures can have a simultaneous effect on multiple systems.
- Divergent impacts of IROFS. Assurance must be provided that the negative impacts of an IROFS, if any, do not outweigh the positive impacts; i.e., to ensure that the application of an IROFS for one safety function does not degrade the defense-in-depth of an unrelated safety function.
- Other safety and mitigating factors that do not achieve the status of IROFS that could impact system performance.
- Identification of scenarios, events, or event sequences with multiple impacts, i.e. impacts on chemical safety, fire safety, criticality safety, and/or radiation safety. For example, a flood might cause both a loss of containment and moderation impacts.
- Potential interactions between processes, systems, areas, and buildings; any interdependence of systems, or potential transfer of energy or materials.
- Major hazards or events, which tend to be common cause situations leading to interactions between processes, systems, buildings, etc.

3.1.2 Process Hazard Analysis Method

As noted above, the HAZOP method was used to identify the process hazards. The HAZOP process hazard analysis (PHA) method is consistent with the guidance provided in NUREG-1513 (NRC, 2001a). Implementation of the HAZOP method was accomplished by either validating the Urenco HAZOPs for the NEF design or performing a new HAZOP for systems where there were no existing HAZOPs. In general, new HAZOPs were performed for the Technical Services Building (TSB) systems. In cases for which there was an existing HAZOP, the ISA Team, through the validation process, developed a new HAZOP.

For the UF_6 process systems, this portion of the ISA was a validation of the HAZOPs provided by Urenco. The validation process involved workshop meetings with the ISA Team. In the workshop meeting, the ISA Team challenged the results of the Urenco HAZOPs. As necessary the HAZOPs were revised/updated to be consistent with the requirements identified in 10 CFR 70 (CFR, 2003b) and as further described in NUREG-1513 (NRC, 2001a) and NUREG-1520 (NRC, 2002a).

To validate the Urenco HAZOPs, the ISA Team performed the following tasks:

- The Urenco process engineer described the salient points of the process system covered by the HAZOP being validated.
- The ISA Team divided the process "Nodes" into reasonable functional blocks.
- The process engineer described the salient points of the items covered by the "Node" being reviewed.
- The ISA Team reviewed the "Guideword" used in the Urenco HAZOP to determine if the HAZOP is likely to identify all credible hazards. A representative list of the guidewords used by the ISA Team is provided in Table 3.1-1, HAZOP Guidewords, to ensure that a complete assessment was performed.
- The ISA Team Leader introduced each Guideword being considered in the ISA HAZOP and the team reviewed and considered the potential hazards.
- For each potential hazard, the ISA Team considered the causes, including potential interactions among materials. Then, for each cause, the ISA Team considered the consequences and consequence severity category for the consequences of interest (Criticality Events, Chemical Releases, Radiation Exposure, Environment impacts). A statement of "No Safety Issue" was noted in the system HAZOP table for consequences of no interest such as maintenance problems or industrial personnel accidents.
- For each hazard, the ISA Team considered existing safeguards designed to prevent the hazard from occurring.
- For each hazard, the ISA Team also considered any existing design features that could mitigate/reduce the consequences.
- The Urenco HAZOP was modified to reflect the ISA Team's input in the areas of hazards, causes, consequences, safeguards and mitigating features.
- For each external event hazard, the ISA Team determined if the external hazard is credible (i.e., external event initiating frequency $>10^{-6}$ per year).

- When all of the Guidewords had been considered for a particular node, the ISA Team applied the same process and guidewords to the next node until the entire process system was completed.

The same process as above was followed for the TSB systems, except that instead of using the validation process, the ISA Team developed a completely new HAZOP. This HAZOP was then used as the hazard identification input into the remainder of the process.

The results of the ISA Team workshops are summarized in the ISA HAZOP Table, which forms the basis of the hazards portion of the Hazard and Risk Determination Analysis. The HAZOP tables are contained in the ISA documentation. The format for this table, which has spaces for describing the node under consideration and the date of the workshop, is provided in Table 3.1-2, ISA HAZOP Table Sample Format. This table is divided into 7 columns:

GUIDEWORD	Identifies the Guideword under consideration.
HAZARD	Identifies any issues that are raised.
CAUSES	Lists any and all causes of the hazard noted.
CONSEQUENCES	Identifies the potential and worst case consequence and consequences severity category if the hazard goes uncontrolled.
SAFEGUARDS	Identifies the engineered and/or administrative protection designed to prevent the hazard from occurring.
MITIGATION	Identifies any protection, engineered or otherwise, that can mitigate/reduce the consequences.
COMMENTS	Notes any comments and any actions requiring resolution.

This approach was used for all of the process system hazard identifications. The “Fire” and “External Events” guidewords were handled as a facility-wide assessment and were not explicitly covered in each system hazard evaluation.

The results of the HAZOP are used directly as input to the risk matrix development.

3.1.3 Risk Matrix Development

3.1.3.1 Consequence Analysis Method

10 CFR 70.61 (CFR, 2003c) specifies two categories for accident sequence consequences: “high consequences” and “intermediate consequences.” Implicitly there is a third category for accidents that produce consequences less than “intermediate.” These are referred to as “low consequence” accident sequences. The primary purpose of PHA is to identify all uncontrolled and unmitigated accident sequences. These accident sequences are then categorized into one of the three consequence categories (high, intermediate, low) based on their forecast radiological, chemical, and/or environmental impacts.

For evaluating the magnitude of the accident consequences, calculations were performed using the methodology described in the ISA documentation. Because the consequences of concern are the chemotoxic exposure to hydrogen fluoride (HF) and UO_2F_2, the dispersion methodology discussed in Section 6.3.2 was used. The dose consequences for all of the accident sequences were evaluated and compared to the criteria for "high" and "intermediate" consequences. The inventory of uranic material for each accident considered was dependent on the specific accident sequence. For criticality accidents, the consequences were conservatively assumed to be high for both the public and workers.

Table 3.1-3, Consequence Severity Categories Based on 10 CFR 70.61, presents the radiological and chemical consequence severity limits of 10 CFR 70.61 (CFR, 2003c) for each of the three accident consequence categories. Table 3.1-4, Chemical Dose Information, provides information on the chemical dose limits specific to the NEF.

3.1.3.2 Likelihood Evaluation Method

10 CFR 70.61 (CFR, 2003c) also specifies the permissible likelihood of occurrence of accident sequences of different consequences. "High consequence" accident sequences must be "highly unlikely" and "intermediate consequence" accident sequences must be "unlikely." Implicitly, accidents in the "low consequence" category can have a likelihood of occurrence less than "unlikely" or simply "not unlikely." Table 3.1-5, Likelihood Categories Based on 10 CFR 70.61, shows the likelihood of occurrence limits of 10 CFR 70.61 (CFR, 2003c) for each of the three likelihood categories.

The definitions of "not unlikely" and "unlikely" are taken from NUREG-1520 (NRC, 2002a). The definition of "highly unlikely" is taken from NUREG-1520 (NRC, 2002a). Additionally, a qualitative determination of "highly unlikely" can apply to passive design component features (e.g., tanks, piping, cylinders, etc.) of the facility that do not rely on human interface to perform the criticality safety function (i.e., termed "safe-by-design"). Safe-by-design components are those components that by their physical size or arrangement have been shown to have a $k_{eff} < 0.95$. The definition of safe-by-design components encompasses two different categories of components. The first category includes those components that are safe-by-volume, safe-by-diameter or safe-by-slab thickness. A set of generic conservative criticality calculations has determined the maximum volume, diameter, or slab thickness (i.e., safe value) that would result in a $k_{eff} < 0.95$. A component in this category has a volume, diameter or slab thickness that is less than the associated safe value resulting from the generic conservative criticality calculations and therefore the k_{eff} associated with this component is < 0.95. The components in the second category require a more detailed criticality analysis (i.e., a criticality analysis of the physical arrangement of the component's design configuration) to show that k_{eff} is < 0.95. In the second category of components, the design configuration is not bounded by the results of the generic conservative criticality calculations for maximum volume, diameter, or slab thickness that would result in a $k_{eff} < 0.95$. Examples of components in this second category are the product pumps that have volumes greater than the safe-by-volume value, but are shown by specific criticality analysis to have a $k_{eff} < 0.95$.

For failure of passive safe-by-design components to be considered "highly unlikely," these components must also meet the criterion that the only potential means to effect a change that might result in a failure to function, would be to implement a design change (i.e., geometry deformation as a result of a credible process deviation or event does not adversely impact the performance of the safety function). The evaluation of the potential to adversely impact the safety function of these passive design features includes consideration of potential mechanisms to cause bulging, corrosion, and breach of confinement/leakage and subsequent accumulation of material. The evaluation further includes consideration of adequate controls to ensure that the double contingency principle is met. For each of these passive design components, it must be concluded, that there is no credible means to effect a geometry change that might result in a failure of the safety function and that significant margin exists. For components that are safe-by-volume, safe-by-diameter, or safe-by-slab thickness (i.e., first category of safe-by-design components), significant margin is defined as a margin of at least 10%, during both normal and upset conditions, between the actual design parameter value of the component and the value of the corresponding critical design attribute. For components that require a more detailed criticality analysis (i.e., second category of safe-by-design components), significant margin is defined as $k_{eff} < 0.95$, where $k_{eff} = k_{calc} + 3\sigma_{calc}$. This margin is considered acceptable since the calculation of k_{eff} also conservatively assumes the components are full of uranic breakdown material at maximum enrichment, the worst credible moderation conditions exist, and the worst credible reflection conditions exist. In addition, the configuration management system required by 10 CFR 70.72 (implemented by the NEF Configuration Management Program) ensures the maintenance of the safety function of these features and assures compliance with the double contingency principle, as well as the defense-in-depth criterion of 10 CFR 70.64(b).

The definition of "not credible" is also taken from NUREG-1520 (NRC, 2002a). If an event is not credible, IROFS are not required to prevent or mitigate the event. The fact that an event is not "credible" must not depend on any facility feature that could credibly fail to function. One cannot claim that a process does not need IROFS because it is "not credible" due to characteristics provided by IROFS. The implication of "credible" in 10 CFR 70.61 (CFR, 2003c) is that events that are not "credible" may be neglected.

Any one of the following independent acceptable sets of qualities could define an event as not credible:

a. An external event for which the frequency of occurrence can conservatively be estimated as less than once in a million years

b. A process deviation that consists of a sequence of many unlikely human actions or errors for which there is no reason or motive (In determining that there is no reason for such actions, a wide range of possible motives, short of intent to cause harm, must be considered. Necessarily, no such sequence of events can ever have actually happened in any fuel cycle facility.)

c. Process deviations for which there is a convincing argument, given physical laws that they are not possible, or are unquestionably extremely unlikely.

3.1.3.3 Risk Matrix

The three categories of consequence and likelihood can be displayed as a 3 x 3 risk index matrix. By assigning a number to each category of consequence and likelihood, a qualitative risk index can be calculated for each combination of consequence and likelihood. The risk index equals the product of the integers assigned to the respective consequence and likelihood categories. The risk index matrix, along with computed risk index values, is illustrated in Table 3.1-6, Risk Matrix with Risk Index Values. The shaded blocks identify accidents of which the consequences and likelihoods yield an unacceptable risk index and for which IROFS must be applied.

The risk indices can initially be used to examine whether the consequences of an uncontrolled and unmitigated accident sequence (i.e., without any IROFS) could exceed the performance requirements of 10 CFR 70.61 (CFR, 2003c). If the performance requirements could be exceeded, IROFS are designated to prevent the accident or to mitigate its consequences to an acceptable level. A risk index value less than or equal to four means the accident sequence is acceptably protected and/or mitigated. If the risk index of an uncontrolled and unmitigated accident sequence exceeds four, the likelihood of the accident must be reduced through designation of IROFS. In this risk index method, the likelihood index for the uncontrolled and unmitigated accident sequence is adjusted by adding a score corresponding to the type and number of IROFS that have been designated.

3.1.4 Risk Index Evaluation Summary

The results of the ISA are summarized in tabular form. This table includes the accident sequences identified for this facility. The accident sequences were not grouped as a single accident type but instead were listed individually in the table. The Table has columns for the initiating event and for IROFS. IROFS may be mitigative or preventive. Mitigative IROFS are measures that reduce the consequences of an accident. The phrase "uncontrolled and/or unmitigated consequences" describes the results when the system of existing preventive IROFS fails and existing mitigation also fails. Mitigated consequences result when the preventive IROFS fail, but mitigative measures succeed. Index numbers are assigned to initiating events, IROFS failure events, and mitigation failure events, based on the reliability characteristics of these items.

With redundant IROFS and in certain other cases, there are sequences in which an initiating event places the system in a vulnerable state. While the system is in this vulnerable state, an IROFS must fail for the accident to result. Thus, the frequency of the accident depends on the frequency of the first event, the duration of vulnerability, and the frequency of the second IROFS failure. For this reason, the duration of the vulnerable state is considered, and a duration index is assigned. The values of all index numbers for a sequence, depending on the number of events involved, are added to obtain a total likelihood index, T. Accident sequences are then assigned to one of the three likelihood categories of the risk matrix, depending on the value of this index in accordance with Table 3.1-8, Determination of Likelihood Category.

The values of index numbers in accident sequences are assigned considering the criteria in Tables 3.1-9 through 3.1-11. Each table applies to a different type of event. Table 3.1-9, Failure Frequency Index Numbers, applies to events that have *frequencies* of occurrence, such as initiating events and certain IROFS failures. Failure Probability Index Numbers are evaluated based on operating experience, (either from Urenco or the National Enrichment Facility, as appropriate) or analyses. When failure *probabilities* are required for an event, Table 3.1-10, Failure Probability Index Numbers, provides the index values. Table 3.1-11, Failure Duration Index Numbers, provides index numbers *for durations* of failure. These are used in certain accident sequences where two IROFS must simultaneously be in a failed state. In this case, one of the two controlled parameters will fail first. It is then necessary to consider the duration that the system remains vulnerable to failure of the second. This period of vulnerability can be terminated in several ways. The first failure may be "fail-safe" or be continuously monitored, thus alerting the operator when it fails so that the system may be quickly placed in a safe state. Or the IROFS may be subject to periodic surveillance tests for hidden failures. When hidden failures are possible, these surveillance intervals limit the duration that the system is in a vulnerable state. The reverse sequences, where the second IROFS fails first, should be considered as a separate accident sequence. This is necessary because the failure frequency and the duration of outage of the first and the second IROFS may differ. The values of these duration indices are not merely judgmental. They are directly related to the time intervals used for surveillance and the time needed to render the system safe.

The duration of failure is accounted for in establishing the overall likelihood that an accident sequence will continue to the defined consequence. Thus, the time to discover and repair the failure is accounted for in establishing the risk of the postulated accident.

The total likelihood index is the sum of the indices for all the events in the sequence, including those for duration. Consequences are assigned to one of the three consequence categories of the risk matrix, based on calculations or estimates of the actual consequences of the accident sequence. The consequence categories are based on the levels identified in 10 CFR 70.61 (CFR, 2003c). Multiple types of consequences can result from the same event. The consequence category is chosen for the most severe consequence.

In summarizing the ISA results, Table 3.7-1, Accident Sequence and Risk Index, provides two risk indices for each accident sequence to permit evaluation of the risk significance of the IROFS involved. To measure whether an IROFS has high risk significance, the table provides an "uncontrolled risk index," determined by modeling the sequence with all IROFS as failed (i.e., not contributing to a lower likelihood). In addition, a "controlled risk index" is also calculated, taking credit for the low likelihood and duration of IROFS failures. When an accident sequence has an uncontrolled risk index exceeding four but a controlled risk index of less than four, the IROFS involved have a high risk significance because they are relied on to achieve acceptable safety performance. Thus, use of these indices permits evaluation of the possible benefit of improving IROFS and also whether a relaxation may be acceptable.

3.2 INTEGRATED SAFETY ANALYSIS TEAM

There were two ISA Teams that were employed in the ISA. The first team worked on the non-classified portions of the facility and is referred to in the text as the ISA Team. The second team, referred to as the Classified ISA Team, performed the ISA on the classified elements of the facility. Both teams were selected with credentials consistent with the requirements in 10 CFR 70.65 (CFR, 2003a) and the guidance provided in NUREG-1520 (NRC, 2002a). To facilitate consistency of results, common membership was dictated as demonstrated below (i.e., some members of the Non-Classified Team participated on the Classified Team. One of the members of the Classified Team participated in the ISA Team Leader Training, which was conducted prior to initiating the ISA. In addition, the Classified ISA Team Leader observed some of the non-classified ISA Team meetings.

The ISA was performed by a team with expertise in engineering, safety analysis and enrichment process operations. The team included personnel with experience and knowledge specific to each process or system being evaluated. The team was comprised of individuals who have experience, individually or collectively, in:

- Nuclear criticality safety
- Radiological safety
- Fire safety
- Chemical process safety
- Operations and maintenance
- ISA methods.

The ISA team leader was trained and knowledgeable in the ISA method(s) chosen for the hazard and accidents evaluations. Collectively, the team had an understanding of all process operations and hazards under evaluation.

The ISA Manager was responsible for the overall direction of the ISA. The process expertise was provided by the Urenco personnel on the team. In addition, the Team Leader has an adequate understanding of the process operations and hazards evaluated in the ISA, but is not the responsible cognizant engineer or enrichment process expert.

3.3 COMPLIANCE ITEM COMMITMENTS

3.3.1 For accident sequences PT3-5, PB1-3, FR1-1, FR1-2, FR2-1, FR2-2, DS1-1, DS1-2, DS2-1, DS2-2, DS3-1, DS3-2, SW1-1, SW1-2, LW1-2, LW1-3, RD1-1, and EC3-1, an Initiating Event Frequency (IEF) index number of "-2" may be assigned based on evidence from the operating history of similar designed Urenco European plants. Detailed justifications for the IEF index numbers of "-2" will be developed during detailed design. If the detailed justification does not support the IEF index number of "-2," then the IEF index number assigned and the associated accident sequence(s) will be re-evaluated and revised, as necessary, consistent with overall ISA methodology.

3.3.2 For Administrative Control IROFS that involve "use of" a component or device, a Failure Probability Index Number (FPIN) of "-2" may be assigned provided the IROFS is a routine, simple, action that either: (1) involves only one or two decision points or (2) is highly detailed in the associated implementing procedure. Alternately, an FPIN of "-3" may be assigned for this type of IROFS provided the criteria specified above for an FPIN of "-2" are met and the IROFS is enhanced by requiring independent verification of the safety function. This enhancement shall meet the requirements for independent verification identified in item 3.3.5 below. If these criteria cannot be met, then the FPIN assigned to the IROFS and the associated accident sequence(s) will be re-evaluated and revised, as necessary, consistent with the overall ISA methodology.

3.3.3 For Administrative Control IROFS that involve "verification of" a state or condition, an FPIN of "-2" may be assigned provided the IROFS is a routine action performed by one person, with proceduralized, objective, acceptance criteria. Alternately, an FPIN of "-3" may be assigned for this type of IROFS provided the criteria specified above for an FPIN of "-2" are met and the IROFS is enhanced by requiring independent verification of the safety function. This enhancement shall meet the requirements for independent verification identified in item 3.3.5 below. If these criteria cannot be met, then the FPIN assigned to the IROFS and the associated accident sequence(s) will be re-evaluated and revised, as necessary, consistent with the overall ISA methodology.

3.3.4 For Administrative Control IROFS that involve " independent sampling," different samples are obtained and an FPIN of "-2" may be assigned provided at least three of the following four criteria are met.

1. Different methods/techniques are used for sample analysis.
2. Samples are obtained from different locations.
3. Samples are obtained at different times. The time period between collection of the different samples shall be sufficient to ensure results are meaningful and representative of the material sampled.
4. Samples are obtained by different personnel.

If at least three of the above criteria cannot be met, then the FPIN assigned to the IROFS and the associated accident sequence(s) will be re-evaluated and revised, as necessary, consistent with the overall ISA methodology.

3.3.5 For IROFS and IROFS with Enhanced Failure Probability Index Numbers (i.e., enhanced IROFS) that require "independent verification" of a safety function, the independent verification shall be independent with respect to personnel and personnel interface. Specifically, a second qualified individual, operating independently (e.g., not at the same time or not at the same location) of the individual assigned the responsibility to perform the required task, shall, as applicable, verify that the required task (i.e., safety function) has been performed correctly (e.g., verify a condition), or re-perform the task (i.e., safety function), and confirm acceptable results before additional action(s) can be taken which potentially negatively impact the safety function of the IROFS. The required task and independent verification shall be implemented by procedure and documented by initials or signatures of the individuals responsible for each task. In addition, the individuals performing the tasks shall be qualified to perform, for the particular system or process (as applicable) involved, the tasks required and shall possess operating knowledge of the particular system or process (as applicable) involved and its relationship to facility safety. The requirements for independent verification are consistent with the applicable guidance provided in ANSI/ANS-3.2-1994, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants.

3.3.6 Upon completion of the design of IROFS, the IROFS boundaries will be defined. In defining the boundaries for each IROFS, Louisiana Energy Services procedure DP-ISA-1.1, "IROFS Boundary Definition," will be used. This procedure requires the identification of each support system and component necessary to ensure the IROFS is capable of performing its specified safety function.

3.3.7 The applicable guidance of the following industry standards, guidance documents and regulatory guides shall be used for the design, procurement, installation, testing, and maintenance of IROFS at the NEF.

a. Institute of Electrical and Electronics Engineers (IEEE) standard IEEE 603-1998, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations"

b. IEEE standard 384-1992, "IEEE Standard Criteria for Independence of Class IE Equipment and Circuits"

c. Branch Technical Position HICB-11, "Guidance on Application and Qualification of Isolation Devices," Revision 4, June 1977, from NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"

d. Regulatory Guide 1.75, "Physical Independence of Electric Systems," Revision 2, September 1978

e. IEEE standard 344-1987, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations"

f. Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," Revision 2, June 1988

g. American National Standards Institute (ANSI)/Instrumentation, Systems, and Automation Society (ISA)-S67.04-1994, Part 1, "Setpoints for Nuclear Safety-Related Instrumentation"

h. Regulatory Guide 3.17, "Earthquake Instrumentation for Fuel Reprocessing Plants," February 1974 (for IROFS26 only)

i. IEEE standard 338-1987, "IEEE Standard Criteria for Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems"

j. Branch Technical Position HICB-17, "Guidance on Self-Test and Surveillance Test Provisions," Revision 4, June 1977, from NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"

k. Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems," Revision 3, April 1995

l. IEEE standard 518-1982, "IEEE Guide for Installation of Electrical Equipment to Minimize Electrical Noise Inputs to Controllers from External Sources"

m. IEEE standard 1050-1996, "IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations"

n. IEEE standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations" (for separation and isolation)

3.3.8 The actual seismic design detailed approach for NEF IROFS will be based on the DOE–STD-1020-2002 (DOE, 2002) or the ASCE Standard Seismic Design Criteria (ASCE, 2003) method and finalized prior to detailed design.

3.3.9 To support the final design of the NEF, additional soil borings will be collected from the NEF site. Laboratory testing will be performed on soil samples and additional in-situ testing will be performed to determine static and dynamic soil properties. Using the soil information obtained, the following activities will be conducted.

- The assessment of soil liquefaction potential will be performed using the applicable guidance of Regulatory Guide 1.198, Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites, dated November 2003 (NRC, 2003a).
- Allowable bearing pressures provided in the ISA Summary will be confirmed using the applicable methods of Naval Facilities Engineering Command Design Manual NAVFAC DM-7.02, Foundations and Earth Structures, dated 1986 (NAVFAC, 1986a); Foundation Engineering Handbook, H.F. Winterkorn and H.Y. Fang, dated 1975 (Winterkorn, 1975); and Foundation Analysis and Design, J.E. Bowles, dated 1996 (Bowles, 1996).
- Building settlement analysis will be performed using the applicable methods of NAVFAC DM-7.01, Soil Mechanics, dated 1986 (NAVFAC, 1986b); and Foundation Engineering Handbook, H.F. Winterkorn and H.Y. Fang, dated 1975 (Winterkorn, 1975). The acceptance criteria for the building settlement analysis will be based on Urenco design criteria for allowable total and differential settlement of equipment and buildings.

3.3.10 The chemical traps on the second floor of the Process Services Area contain hazardous materials and are housed in fire rated enclosures to meet the requirements of Section 6.4 of NFPA 101 (NFPA, 1997).

3.3.11 The Separations Building Modules are designed to meet the occupant and exiting requirements set by NFPA 101 (NFPA, 1997) and to meet the construction type classifications set by the New Mexico Building Code (NMBC, 1997).

3.3.12 The floors of the Cascade Halls have a floor profile quality classification of flat in accordance with ACI 117-90 (ACI, 1990a) to aid in the transport of assembled centrifuges.

3.3.13 The Technical Services Building is designed to meet the occupant and exiting requirements set by the
NFPA 101 (NFPA, 1997) and to meet the construction type classifications set by the New Mexico Building Code (NMBC, 1997).

3.3.14 The Cylinder Receipt and Dispatch Building is designed to meet the occupant and exiting requirements set by the
NFPA 101 (NFPA, 1997) and to meet the construction type classification set by the New Mexico Building Code (NMBC, 1997).

3.3.15 The Centrifuge Assembly Building (CAB) is designed to meet the occupant and exiting requirements set by NFPA 101 (NFPA, 1997) and to meet the construction type classifications set by the New Mexico Building code (NMBC, 1997) and as Type I Construction by NFPA 220 (NFPA, 1999).

3.3.16 Centrifuge assembly activities are undertaken in clean room conditions, ISO Class 5 according to ISO 14644-1:1999E (ISO, 1999), to prevent ingress of volatile contaminants which would have a detrimental effect on centrifuge performance.

3.3.17 The floors of the CAB Assembled Centrifuge Storage Area have a floor profile quality classification of flat in accordance with ACI 117-90 (ACI, 1990a) to aid in the transport of assembled centrifuges.

3.3.18 The Blending and Liquid Sampling Area is designed to meet the occupant and exiting requirements set by the NFPA 101 (NFPA, 1997) and to meet the construction type classification set by the New Mexico Building Code (NMBC, 1997).

3.3.19 The Central Utilities Building is designed to meet the occupant and exiting requirements set by the NFPA 101 (NFPA, 1997) and set by the New Mexico Building Code (NMBC, 1997).

3.3.20 The Administration Building is designed to meet the occupant and exiting requirements set by the NFPA 101 (NFPA, 1997) and by the New Mexico Building Code (NMBC, 1997).

3.3.21 These buildings are designed to meet the occupant and exiting requirements set by the NFPA 101 (NFPA, 1997) and the construction type classifications set by the New Mexico Building Code (NMBC, 1997).

3.3.22 The following codes and standards are generally applicable to the structural design of the National Enrichment Facility:

- New Mexico Building Code (NMBC, 1997)
- Uniform Building Code (UBC, 1997)
- ASCE 7-98, Minimum Design Loads for Buildings and Other Structures (ASCE, 1998)
- ACI 318-99, Building Code Requirements for Structural Concrete (ACI, 1999)

- ACI 349-90, Code Requirements for Nuclear Safety Related Concrete Structures (ACI, 1990b)
- AISC Manual of Steel Construction, Ninth Edition (AISC, 1989)
- PCI Design Handbook, Fifth Edition (PCI, 1999)
- American Society of Testing and Materials (ASTM).

3.3.23 Structural Design Loads

a. The determination of wind pressure loadings and the design for wind loads for all safety significant structures and components exposed to wind are based on the requirements of ASCE 7-98 (ASCE, 1998). The determination of wind pressure loadings and the design for wind loads for all other structures and components exposed to wind are based on the requirements of the Uniform Building Code (UBC, 1997), Chapter 16 which further refers to the wind design requirements of ASCE 7-98, Section 6.0 (ASCE, 1998).

b. For reinforced concrete targets, the formulas used to establish the missile depth of penetration (x) and scabbing thickness (t_s) are based on the Modified National Defense Research Committee Formula (NDRC) (ASCE, 1980) and the Army Corps of Engineers Formula (ACE) (ASCE, 1980) respectively.

c. Per Section C.7.2.2 of ACI 349-90 (ACI, 1990b), the concrete thickness required to resist hard missiles shall be at least 1.2 times the scabbing thickness, t_s. Punching shear is calculated and checked against the requirements of ACI 349-90 (ACI, 1990b), Section C.7.2.3.

d. For steel targets, the formula used to establish the perforation thickness is the Ballistic Research Laboratory (BRL) Formula (ASCE, 1980).

e. All buildings and structures, including such items as equipment supports, are designed to withstand the earthquake loads defined in Chapter 16, Division IV of the Uniform Building Code (UBC, 1997).

f. Snow loadings on roofs and other exposed surfaces for non-safety significant structures are determined in accordance with the Uniform Building Code (UBC, 1997), Chapter 16, Division II.

g. Load combinations for concrete structures and components for the safety significant structures are based on ACI 349-90 (ACI, 1990b). Load combinations for other concrete structures are based on ASCE 7-98 (ASCE, 1998). All concrete structures are designed using the ACI Strength Design Method (ACI, 1999).

h. Load combinations for steel structures and components for all buildings are based on ASCE 7-98 (ASCE, 1998). All structural steel is designed using the AISC Allowable Stress Method (AISC, 1989).

i. Design live loads, including impact loads, used are in accordance with Section 4.0 and Table 4-1 of ASCE 7-98 (ASCE, 1998).

j. During detailed design of specific buildings and areas, pressure loads due to postulated truck and pipeline explosions will be considered. The pressure loads will be developed in accordance with the underlying assumptions used in the explosion hazard assessments described in Sections 3.2.1.2.1 and 3.2.2.4 of the

ISA Summary. These buildings and areas include: Separations Building Modules (UF_6 Handling Area, Process Services Area and Cascade Halls), Blending and Liquid Sampling Area, Cylinder Receipt and Dispatch Building, Technical Services Building and the Centrifuge Test Facility. These buildings and areas are constructed of concrete.

3.3.24 Natural UF_6 feed is received at the NEF in Department of Transportation (DOT) 7A, Type A cylinders from a conversion plant. The cylinders are ANSI N14.1 (ANSI, applicable version), 48Y or 48X cylinders.

3.3.25 Applicable codes and standards for process systems are reflected in Tables 3.3-1 through 3.3-7.

3.3.26 Product Liquid Sampling Autoclave

a. The pressure vessel is designed and fabricated in accordance with the requirements of ASME Section VIII, Division1 (current version at the time of autoclave manufacture), with the exception that the pressure relief devices specified in Sections UG-125 through 137 are not be provided due to the potential for release of hazardous material to the environment through a pressure relief device. Instead, two independent and diverse automatic trips of the autoclave heaters and fan motor are provided to eliminate the heat input and preclude approaching the autoclave design pressure. This is considered to be acceptable due to the large margin between the autoclave design pressure 12 bar (174 psia) and the maximum allowable working pressure 1.8 bar (26 psia) and the fail-safe design of the two independent and diverse automatic trips of the autoclave heaters and fan motor. The pressure vessel is also tested and stamped to the requirements of ASME Section VIII, Division 1 rules and is registered with the National Board.

b. The autoclave is designed and tested to ensure leak tight integrity is maintained.

c. The autoclave door seal is leak tested and inspected prior to each autoclave sample sequence.

3.3.27 Separations Building Gaseous Effluent Vent System (GEVS)

a. The Separations Building GEVS provides for continuous monitoring and periodic sampling of the gaseous effluent in the exhaust stack in accordance with the guidance in Regulatory Guide 4.16 (NRC, 1985).

b. The design and in-place testing of the Separations Building GEVS will be consistent with the applicable guidance in Regulatory Guide 1.140 (NRC, 2001b), ASME AG-1-1997 (ASME, 1997), and ASME N510-1989 (ASME, 1989). The system includes potassium carbonate impregnated activated charcoal filters for HF removal. As such, the portions of Regulatory Guide 1.140 (NRC, 2001b), ASME AG-1-1997 (ASME, 1997), and ASME N510-1989 (ASME, 1989), which address activated charcoal filters for radioiodine removal are not applicable. The prefilter efficiency (85%) is based on testing in accordance with ASME AG-1-1997 (ASME, 1997). The HEPA filter efficiency (99.97%) is based on removal of 0.3 micron particles when tested in accordance with ASME-AG-1 (ASME, 1997). The impregnated charcoal filter efficiency (99%) for removal of HF is based on Urenco specifications. In-place testing and inspections of the filters will be performed in accordance with the guidance in Regulatory Guidance 1.140 (NRC, 2001b). The frequency for performance of in-place filter testing and the acceptance criteria for penetration and leakage (or bypass) will be consistent with the guidance in Regulatory Guide 1.140 (NRC, 2001b). Qualification testing, to verify HF removal efficiency, of the impregnated charcoal will be performed using ASTM D6646-03 (ASTM, 2003), modified to reflect removal of HF instead of hydrogen sulfide. Laboratory testing of the impregnated charcoal filter of charcoal samples will be performed on an annual basis. Throughout the useful life of the impregnated charcoal, the impregnate is progressively consumed. The laboratory testing will determine the impregnant content within the sample. The amount of impregnant present in the sample is indicative of the remaining life of charcoal bed for removal of HF.

3.3.28 Technical Support Building (TSB) GEVS

a. The TSB GEVS provides for continuous monitoring and periodic sampling of the gaseous effluent in the exhaust stack in accordance with the guidance in Regulatory Guide 4.16 (NRC, 1985).

b. The design and in-place testing of the TSB GEVS will be consistent with the applicable guidance in Regulatory Guide 1.140 (NRC, 2001b), ASME AG-1-1997 (ASME, 1997), and ASME N510-1989 (ASME, 1989). The system includes a potassium carbonate impregnated activated charcoal filter for HF removal. As such, the portions of Regulatory Guide 1.140 (NRC, 2001b), ASME AG-1-1997 (ASME, 1997), and ASME N510-1989 (ASME, 1989), which address activated charcoal filters for radioiodine removal are not applicable. The prefilter efficiency (85%) is based on testing in accordance with ASME AG-1-1997 (ASME, 1997). The HEPA filter efficiency (99.97%) is based on removal of 0.3 micron particles when tested in accordance with ASME-AG-1 (ASME, 1997). The impregnated charcoal filter efficiency (99%) for removal of HF is based on Urenco specifications. In-place testing and inspections of the filters will be performed in accordance with the guidance in Regulatory Guidance 1.140 (NRC, 2001b). The frequency for performance of in-place filter testing and the acceptance criteria for penetration and leakage (or bypass) will be consistent with the guidance in Regulatory Guide 1.140 (NRC, 2001b). Qualification testing, to verify HF removal efficiency, of the impregnated charcoal will be performed using ASTM D6646-03 (ASTM, 2003), modified to reflect removal of HF instead of hydrogen sulfide. Laboratory testing of the impregnated charcoal filter of charcoal samples will be performed on an annual basis. Throughout the useful life of the impregnated charcoal, the impregnate is progressively consumed. The laboratory testing will determine the impregnant content within the sample. The amount of impregnant present in the sample is indicative of the remaining life of charcoal bed for removal of HF.

3.3.29 Centrifuge Test and Post Mortem Facilities Exhaust Filtration System

a. The Centrifuge Test and Post Mortem Facilities Exhaust Filtration System provides for continuous monitoring and periodic sampling of the gaseous effluent in the exhaust stack in accordance with the guidance in Regulatory Guide 4.16 (NRC, 1985).

b. The design and in-place testing of the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System will be consistent with the applicable guidance in

Regulatory Guide 1.140 (NRC, 2001b), ASME AG-1-1997 (ASME, 1997), and ASME N510-1989 (ASME, 1989). The system includes a potassium carbonate impregnated activated charcoal filter for HF removal. As such, the portions of Regulatory Guide 1.140 (NRC, 2001b), ASME AG-1-1997 (ASME, 1997), and ASME N510-1989 (ASME, 1989), which address activated charcoal filters for radioiodine removal are not applicable. The prefilter efficiency (85%) is based on testing in accordance with ASME AG-1-1997 (ASME, 1997). The HEPA filter efficiency (99.97%) is based on removal of 0.3 micron particles when tested in accordance with ASME-AG-1 (ASME, 1997). The impregnated charcoal filter efficiency (99%) for removal of HF is based on Urenco specifications. In-place testing and inspections of the filters will be performed in accordance with the guidance in Regulatory Guidance 1.140 (NRC, 2001b). The frequency for performance of in-place filter testing and the acceptance criteria for penetration and leakage (or bypass) will be consistent with the guidance in Regulatory Guide 1.140 (NRC, 2001b). Qualification testing, to verify HF removal efficiency, of the impregnated charcoal will be performed using ASTM D6646-03 (ASTM, 2003), modified to reflect removal of HF instead of hydrogen sulfide. Laboratory testing of the impregnated charcoal filter of charcoal samples will be performed on an annual basis. Throughout the useful life of the impregnated charcoal, the impregnate is progressively consumed. The laboratory testing will determine the impregnant content within the sample. The amount of impregnant present in the sample is indicative of the remaining life of charcoal bed for removal of HF.

3.3.30 In response to Bulletin 2003-03 (NRC, 2003b), LES will not purchase UF_6 cylinders with the 1-in Hunt valves installed nor purchase any replacement 1-in valves from Hunt.

In the unlikely event that any cylinders are received at the NEF with the 1-in Hunt valves installed, the following actions will be taken.

- If the cylinder is empty, the valve will be replaced before the cylinder is used in the facility.
- If the cylinder is filled, a safety justification to support continued use of the cylinder until the valve can be replaced will be developed or the valve will be replaced in accordance with NEF procedures.

No cylinders with the 1-in Hunt valve installed will be used as UBCs.

3.3.31 The containers used for intercontinental shipping are International Organization for Standardization Series 1 freight containers that are supplied in accordance with the ISO 668:1995 (ISO, 1995) Standard.

3.3.32 In the Cylinder Preparation Room, cylinders are pressure tested using compressed air in accordance with ANSI N14-2001 (ANSI, 2001). This system is used for testing new and decontaminated empty cylinders only.

3.3.33 Applicable codes and standards for utility and support systems are reflected in Table 3.3-8.

3.3.34 Exhaust flow from the potentially contaminated rooms (i.e., Ventilated Room, Cylinder Preparation Room and Decontamination Workshop) of the TSB is filtered by a pre-filter, activated carbon filter and HEPA filter and is then released through an exhaust stack.

The exhaust stack flow is continuously monitored for alpha and HF. The stack exhaust is periodically sampled. The continuous monitoring and periodic sampling is in accordance with the guidance in Regulatory Guide 4.16 (NRC, 1985).

3.3.35 The Electrical System design complies with the following codes and standards.

- IEEE C2-2002, National Electrical Safety Code (IEEE, 2002)
- NFPA 70, National Electric Code (NFPA, 1996)
- NFPA 70E, Standard for Electrical Safety Requirements for Employee Workplaces (NFPA, 2000).

3.3.36 The criticality safety for tanks that are not "geometrically safe" or "geometrically favorable" will utilize two independent IROFS for mass control, one IROFS is referred to as "bookkeeping measures" and the second IROFS is referred to as "sampled and analyzed," e.g., tank contents are sampled and analyzed before being transferred to another tank or out of the system. The "bookkeeping measures" is a process to calculate the potential mass of uranium in the tank for any batch operation to ensure that no tank holds more than a safe mass of uranium. This calculated mass of uranium is then compared to a mass limit, which is based on the double-batching limit on mass of uranium in a vessel from the criticality safety analyses. The "bookkeeping measures" process is described in further detail below.

- For NEF, the "bookkeeping measures" are only applied to tanks where the mass of uranium involved, even when double batching error is considered, is far below the safe value. Bookkeeping measures are a documented running inventory estimate of the total uranium mass in a particular tank. The mass inventory for each batch operation is calculated based on the mass of material to be transferred during each batch operation and the mass inventory in the tank prior to the addition of the material from the batch operation.

- There are two types of batch operations that are considered. The first type is liquid transfer between tanks based on moving a volume of liquid with uranic material present in the volume. The second is transferring a number of components into the tank with the uranic material contained within or on the components transferred in each batch operation. For both types of operations, the initial mass inventory is set after emptying, cleaning, and readying the tank for receipt of uranic material. For each batch operation, the amount of uranic material to be transferred during a particular batch operation is estimated. This quantity of material is then credited/debited to/from each tank as appropriate. A new mass inventory in each tank is calculated. The calculated receiving tank mass inventory is compared to the mass limit for the tank prior to the transfer.

- For the second type, a transfer of a number of facility components into an open tank during a batch operation, the mass inventory on/within the components is estimated, and that mass credited to the receiving tank. The final mass inventory in the tank is calculated and the total is compared to the mass limit for the tank prior to the transfer. Open tanks associated with this system are located in the Decontamination Workshop.

3.3.37 UF_6 cylinders with faulty valves are serviced in the Ventilated Room. In the Ventilated Room, the faulty valve is removed and the threaded connection in the cylinder is inspected. A new valve is then installed in accordance with the requirements of ANSI N-14.1 (ANSI, 2001).

3.3.38 IROFS will be designed, constructed, tested and maintained to QA Level 1. IROFS will comply with design requirements established by the ISA and the applicable codes and standards (current approved version at the time of design). IROFS components and their designs will be of proven technology for their intended application. These IROFS components and systems will be qualified to perform their required safety functions under normal and accident conditions, e.g., pressure, temperature, humidity, seismic motion, electromagnetic interference, and radio-frequency interference, as required by the ISA. IROFS components and systems will be qualified using the applicable guidance in Institute of Electrical and Electronics Engineers (IEEE) standard IEEE-323, 1983, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations" (IEEE, 1983). Additionally, non-IROFS components and systems will be qualified to withstand environmental stress caused by environmental and dynamic service conditions under which their failure could prevent satisfactory accomplishment of the IROFS safety functions. Furthermore, IROFS components and systems will be designed, procured, installed, tested, and maintained using the applicable guidance in Regulatory Guide 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," Revision 1, dated October 2003 (NRC, 2003c). IROFS systems will be designed and maintained consistent with the reliability assumptions in the ISA. Redundant IROFS systems will be separate and independent from each other. IROFS systems will be designed to be fail-safe. In addition, IROFS systems will be designed such that process control system failures will not affect the ability of the IROFS systems to perform their required safety functions. Plant control systems will not be used to perform IROFS functions. Installation of IROFS systems will be in accordance with engineering specifications and manufacturer's recommendations. Required testing and calibration of IROFS will be consistent with the assumptions of the ISA and setpoint calculations, as applicable. For hardware IROFS involving instrumentation which provides automatic prevention or mitigation of events, setpoint calculations are performed in accordance with a setpoint methodology, which is consistent with the applicable guidance provided in Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation," Revision 3, dated December 1999 (NRC, 1999).

3.3.39 For those IROFS requiring operator actions, a human factors engineering review of the human-system interfaces shall be conducted using the applicable guidance in NUREG-0700, "Human-System Interface Design Review Guidelines," Revision 2, dated May 2002 (NRC, 2002b), and NUREG-0711, "Human Factors Engineering Program Review Model," Revision 2, dated February 2004 (NRC, 2004).

3.4 REFERENCES

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TABLES

Table 3.1-1 HAZOP Guidewords

UF_6 PROCESS GUIDEWORDS			
Less Heat	Corrosion	Maintenance	No Flow
More Heat	Loss of Services	Criticality	Reverse Flow
Less Pressure	Toxicity	Effluents/Waste	Less Uranium
More Pressure	Contamination	Internal Missile	More Uranium
Impact/Drop	Loss of Containment	Less Flow	Light Gas
Fire (Process, internal, other)	Radiation	More Flow	External Event
NON UF_6 PROCESS GUIDEWORDS			
High Flow	Low Pressure	Impact/Drop	More Uranium
Low Flow	High Temperature	Corrosion	External Event
No Flow	Low Temperature	Loss of Services	Startup
Reverse Flow	Fire	Toxicity	Shutdown
High Level	High Contamination	Radiation	Internal Missile
Low Level	Rupture	Maintenance	
High Pressure	Loss of Containment	Criticality	
No Flow			
EXTERNAL EVENTS POTENTIAL CAUSES			
Construction on Site	Hurricane	Seismic	Transport Hazard Off-Site
Flooding	Industrial Hazard Off-site	Tornado	External Fire
Airplane	Snow/Ice	Local Intense Precipitation	

Table 3.1-2 ISA HAZOP Table Sample Format

Page 1 of 1

ISA HAZOP NODE:		DESCRIPTION :			DATE:	PAGE:
GUIDEWORD	HAZARD	CAUSE	CONSEQUENCE	SAFEGUARDS	MITIGATING FACTORS	COMMENTS

Table 3.1-3 Consequence Severity Categories Based on 10 CFR 70.61
Page 1 of 1

	Workers	Offsite Public	Environment
Category 3 High Consequence	Radiation Dose (RD) >1 Sievert (Sv) (100 rem) For the worker (elsewhere in room), except the worker (local), Chemical Dose (CD) > AEGL-3 For worker (local), CD > AEGL-3 for HF CD > * for U	RD > 0.25 Sv (25 rem) 30 mg sol U intake CD > AEGL-2	–
Category 2 Intermediate Consequence	0.25 Sv (25 rem) <RD≤ 1 Sv (100 rem) For the worker (elsewhere in room), except the worker (local), AEGL-2 < CD≤ AEGL-3 For the worker (local), AEGL-2 < CD ≤ AEGL-3 for HF ** < CD ≤ * for U	0.05 Sv (5 rem) < RD≤ 0.25 Sv (25 rem) AEGL-1 <CD≤ AEGL-2	Radioactive release > 5000 x Table 2 Appendix B of 10 CFR Part 20
Category 1 Low Consequence	Accidents of lower radiological and chemical exposures than those above in this column	Accidents of lower radiological and chemical exposures than those above in this column	Radioactive releases with lower effects than those referenced above in this column

Notes:

*NUREG-1391 threshold value for intake of soluble U resulting in permanent renal failure

**NUREG-1391 threshold value for intake of soluble U resulting in no significant acute effects to an exposed individual

Table 3.1-4 Chemical Dose Information

Page 1 of 1

	High Consequence (Category 3)	Intermediate Consequence (Category 2)
Worker (local)	> 40 mg U intake > 139 mg HF/m^3	> 10 mg U intake > 78 mg HF/m^3
Worker (elsewhere in room)	> 146 mg U/m^3 > 139 mg HF/m^3	> 19 mg U/m^3 > 78 mg HF/m^3
Outside Controlled Area (30-min exposure)	> 13 mg U/m^3 > 28 mg HF/m^3	> 2.4 mg U/m^3 > 0.8 mg HF/m^3

Table 3.1-5 Likelihood Categories Based on 10 CFR 70.61
Page 1 of 1

	Likelihood Category	Probability of Occurrence*
Not Unlikely	3	More than 10^{-4} per-event per-year
Unlikely	2	Between 10^{-4} and 10^{-5} per-event per-year
Highly Unlikely	1	Less than 10^{-5} per-event per-year

*Based on approximate order-of-magnitude ranges

Table 3.1-6 Risk Matrix with Risk Index Values

Severity of Consequences	Likelihood of Occurrence		
	Likelihood Category 1 Highly Unlikely (1)	Likelihood Category 2 Unlikely (2)	Likelihood Category 3 Not Unlikely (3)
Consequence Category 3 High (3)	Acceptable Risk 3	Unacceptable Risk 6	Unacceptable Risk 9
Consequence Category 2 Intermediate (2)	Acceptable Risk 2	Acceptable Risk 4	Unacceptable Risk 6
Consequence Category 1 Low (1)	Acceptable Risk 1	Acceptable Risk 2	Acceptable Risk 3

Table 3.1-7 (Not Used)

Table 3.1-8 Determination of Likelihood Category

Page 1 of 1

Likelihood Category	Likelihood Index T (= sum of index numbers)
1	$T \leq -5$
2	$-5 < T \leq -4$
3	$-4 < T$

Table 3.1-9 Failure Frequency Index Numbers

Page 1 of 2

Frequency Index No.	Based On Evidence	Based On Type Of IROFS**	Comments
-6*	External event with freq. < 10^{-6} /yr		If initiating event, no IROFS needed.
-5*	Initiating event with freq. < 10^{-5}/yr		For passive safe-by-design components or systems, failure is considered highly unlikely when no potential failure mode (e.g., bulging, corrosion, or leakage) exists, as discussed in Section 3.1.3.2, significant margin exists*** and these components and systems have been placed under configuration management.
-4*	No failures in 30 years for hundreds of similar IROFS in industry	Exceptionally robust passive engineered IROFS (PEC), or an inherently safe process, or two independent active engineered IROFS (AECs), PECs, or enhanced admin. IROFS	Rarely can be justified by evidence. Further, most types of single IROFS have been observed to fail
-3*	No failures in 30 years for tens of similar IROFS in industry	A single IROFS with redundant parts, each a PEC or AEC	
-2*	No failure of this type in this facility in 30 years	A single PEC	
-1*	A few failures may occur during facility lifetime	A single AEC, an enhanced admin. IROFS, an admin. IROFS with large margin, or a redundant admin. IROFS	
0	Failures occur every 1 to 3 years	A single administrative IROFS	
1	Several occurrences per year	Frequent event, inadequate IROFS	Not for IROFS, just initiating events

Table 3.1-9 Failure Frequency Index Numbers

Frequency Index No.	Based On Evidence	Based On Type Of IROFS**	Comments
2	Occurs every week or more often	Very frequent event, inadequate IROFS	Not for IROFS, just initiating events

*Indices less than (more negative than) –1 should not be assigned to IROFS unless the configuration management, auditing, and other management measures are of high quality, because, without these measures, the IROFS may be changed or not maintained.

**The index value assigned to an IROFS of a given type in column 3 may be one value higher or lower than the value given in column 1. Criteria justifying assignment of the lower (more negative) value should be given in the narrative describing ISA methods. Exceptions require individual justification.

***For components that are safe-by-volume, safe-by-diameter, or safe-by-slab thickness, significant margin is defined as a margin of at least 10%, during both normal and upset conditions, between the actual design parameter value of the component and the value of the critical design attribute. For components that require a more detailed criticality analysis, significant margin is defined as $k_{eff} < 0.95$, where $k_{eff} = k_{calc} + 3\sigma_{calc}$.

Table 3.1-10 Failure Probability Index Numbers

Page 1 of 1

Probability Index No.	**Probability of Failure on Demand**	**Based on Type of IROFS**	**Comments**
-6*	10^{-6}		If initiating event, no IROFS needed.
-4 or -5*	10^{-4} - 10^{-5}	Exceptionally robust passive engineered IROFS (PEC), or an inherently safe process, or two redundant IROFS more robust than simple admin. IROFS (AEC, PEC, or enhanced admin.)	Can rarely be justified by evidence. Most types of single IROFS have been observed to fail
-3 or -4*	10^{-3} - 10^{-4}	A single passive engineered IROFS (PEC) or an active engineered IROFS (AEC) with high availability	
-2 or -3*	10^{-2} - 10^{-3}	A single active engineered IROFS, or an enhanced admin. IROFS, or an admin. IROFS for routine planned operations	
-1 or -2	10^{-1} - 10^{-2}	An admin. IROFS that must be performed in response to a rare unplanned demand	

*Indices less than (more negative than) –1 should not be assigned to IROFS unless the configuration management, auditing, and other management measures are of high quality, because, without these measures, the IROFS may be changed or not maintained.

Table 3.1-11 Failure Duration Index Numbers

Duration Index No.	Avg. Failure Duration	Duration in Years	Comments
1	More than 3 yrs	10	
0	1 yr	1	
-1	1 mo	0.1	Formal monitoring to justify indices less than -1
-2	A few days	0.01	
-3	8 hrs	0.001	
-4	1 hr	10^{-4}	
-5	5 min	10^{-5}	

Table 3.3-1 Cascade System Codes and Standards

Page 1 of 1

The Centrifuge Machine Passive Isolation Devices is designed, constructed, tested, and maintained to QA Level 1.
Rotating equipment is designed in accordance with the appropriate industry codes and standards.
Heat transfer equipment is designed in accordance with the appropriate industry codes and standards.
All miscellaneous equipment is designed in accordance with the appropriate industry codes and standards.
All process piping in the Cascade System shall meet or exceed the requirements of American Society of Mechanical Engineers, Process Piping, ASME B31.3, current edition at the time of detail engineering.
The design of electrical systems and components in the Cascade System is in conformance with the requirements of the National Electrical Safety Code, IEEE C2, current edition in effect at detail design, and the National Fire Protection Association, National Electrical Code, NFPA 70, current edition in effect at detail engineering, and appropriate industry codes and standards.

Table 3.3-2 Product Take-off System Codes and Standards

Page 1 of 1

The equipment IROFS are designed, constructed, tested, and maintained to QA Level 1.
Rotating equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 rotating equipment in the Product Take-off System.
Heat transfer equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 heat transfer equipment in the Product Take-off System.
Material handling equipment is designed in accordance with the appropriate industry codes and standards and the requirements of the Occupational Safety and Health Administration. There is no QA Level 1 material handling equipment in the Product Take-off System.
All miscellaneous equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 miscellaneous equipment in the Product Take-off System.
All process piping in the Product Take-off System shall meet or exceed the requirements of American Society of Mechanical Engineers, Process Piping, ASME B31.3, current edition at the time of detail design.
All 30-in and 48-in cylinders used in the Product Take-off System comply with the requirements of ANSI N14.1, Uranium Hexafluoride Packaging for Transport, version in effect at the time of cylinder manufacture.

Table 3.3-3 Tails Take-off System Codes and Standards

The equipment IROFS are designed, constructed, tested, and maintained to QA Level 1.
Rotating equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 rotating equipment in the Tails Take-off System.
Heat transfer equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 heat transfer equipment in the Tails Take-off System.
Material handling equipment is designed in accordance with the appropriate industry codes and standards and the requirements of the Occupational Safety and Health Administration. There is no QA Level 1 material handling equipment in the Tails Take-off System.
All miscellaneous equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 miscellaneous equipment in the Tails Take-off System.
All process piping in the Tails Take-off System shall meet or exceed the requirements of American Society of Mechanical Engineers, Process Piping, ASME B31.3, current edition at the time of detail design.
All 48-in cylinders used in the Tails Take-off System comply with the requirements of ANSI N14.1, *Uranium Hexafluoride Packaging for Transport*, version in effect at the time of cylinder manufacture.

Table 3.3-4 Product Blending System Codes and Standards

The equipment IROFS are designed, constructed, tested, and maintained to QA Level 1.
Rotating equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 rotating equipment in the Product Blending System.
Heat transfer equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 heat transfer equipment in the Product Blending System.
Material handling equipment is designed in accordance with the appropriate industry codes and standards and the requirements of the Occupational Safety and Health Administration. There is no QA Level 1 material handling equipment in the Product Blending System.
All miscellaneous equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 miscellaneous equipment in the Product Blending System.
All process piping in the Product Blending System shall meet or exceed the requirements of American Society of Mechanical Engineers, Process Piping, ASME B31.3, current edition.
All 30-in and 48-in cylinders used in the Product Blending System comply with the requirements of ANSI N14.1, Uranium Hexafluoride Packaging for Transport, version in effect at the time of cylinder manufacture.

Table 3.3-5 Product Liquid Sampling System Codes and Standards

The equipment IROFS are designed, constructed, tested, and maintained to QA Level 1.
Product Liquid Sampling Autoclaves and their supports are designed to meet the requirements of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, *Section VIII, Division I, current edition at the time of detail design.*
Rotating equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 rotating equipment in the Product Liquid Sampling System.
Heat transfer equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 heat transfer equipment in the Product Liquid Sampling System.
Material handling equipment is designed in accordance with the appropriate industry codes and standards and the requirements of the Occupational Safety and Health Administration. There is no QA Level 1 material handling equipment in the Product Liquid Sampling System.
All miscellaneous equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 miscellaneous equipment in the Product Liquid Sampling System.
All process piping in the Product Liquid Sampling System shall meet or exceed the requirements of American Society of Mechanical Engineers, Process Piping, ASME B31.3, current edition at the time of detail design.
All 1.5-in and 30-in cylinders used in the Product Liquid Sampling System comply with the requirements of ANSI N14.1, Uranium Hexafluoride Packaging for Transport, version in effect at the time of cylinder manufacture.

Table 3.3-6 Contingency Dump System Codes and Standards

Page 1 of 1

The equipment IROFS are designed, constructed, tested, and maintained to QA Level 1.
Rotating equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 rotating equipment in the Contingency Dump System.
Heat transfer equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 heat transfer equipment in the Contingency Dump System.
All miscellaneous equipment is designed in accordance with the appropriate industry codes and standards. There is no QA Level 1 miscellaneous equipment in the Contingency Dump System.
All process piping in the Contingency Dump System meets or exceeds the requirements of American Society of Mechanical Engineers, Process Piping, ASME B31.3, current edition at the time of detail design.

Table 3.3-7 Gaseous Effluent Vent System Codes and Standards

Equipment Type	Code or Standard
Air Handling Units	NFPA 90A, 1999 AMCA Pub. 99 – 1986 AMCA Pub. 261 – 1998 ARI 430 – 1980 NEMA MG – 1998 REV. 3
Fans/Motors	AMCA 210 – 1999 ASHRAE 51 – 1999 ASHRAE Systems and Equipment 2000 NEMA MG1 – 1998 REV. 3
Coils	ANSI/ARI 410 – 2001
Air Cleaning Devices	ASME AG-1-1997 ERDA 76-21 – 1976 ANSI/ASME N509 – 1989 (R1996) ANSI/ASME N510 – 1989 (R1995) ASME NQA-1 – 2001 ASTM D6646-03 ANSI/AWS-D9.1 – 2000
Dampers	UL-Building Materials Directory

Table 3.3-8 Utility and Support Systems Codes and Standards

ACI 318-99, Building Code Requirements for Structural Concrete, 1999.
ACI 349-90, Code Requirements for Nuclear Safety Related Concrete Structures, 1990.
AIChE, Guidelines for Hazard Evaluation Procedures, 1992.
AISC Manual of Steel Construction – Allowable Stress Design, Ninth Edition, 1989
ANSI N14.1-2001, American National Standard for Nuclear Materials - Uranium Hexafluoride Packaging for Transport, 2001.
ANSI N15.5-1972, Statistical Terminology and Notation for Nuclear Materials Management, 1972.
ASCE 58, Structural Analysis and Design of Nuclear Plant Facilities, Manuals and Reports on Engineering Practice, 1980.
ASCE 7-98, Minimum Design Loads for Building and Other Structures, 1998.
ASME B31.3-2002, Process Piping, 2002.
ASME, Boiler and Pressure Vessel Code, Section VIII, Division 1, 1999.
ASME, NQA-1-1994, Quality Assurance Requirements for Nuclear Facility Applications, 1994.
ASME, NQA-1a-1995, Addenda to ASME NQA-1-1994 Edition, Quality Assurance Requirements for Nuclear Facility Applications, 1995.
ASTM C761-01 - Standard Test Methods for Chemical, Mass Spectrometric, Spectrochemical, Nuclear, and Radiochemical Analysis of Uranium Hexafluoride, 2001.
ASTM E 814, Fire Tests of Through-Penetration Fire Stops.
ERDA 76-21, Nuclear Air Cleaning Handbook, 1976.
IEEE 336, Standard Installation, Inspection, and Testing Requirements for Power, Instrumentation, and Control Equipment at Nuclear Facilities, 1991.
IEEE C2-2002, National Electrical Safety Code, 2002.

Table 3.3-8 Utility and Support Systems Codes and Standards

ISO 668: 1995, Series 1 Freight Containers - Classification, Dimensions and Ratings, 1995.
NFPA 1, Fire Prevention Code, 1997.
NFPA 10, Portable Fire Extinguishers, 1994.
NFPA 101, Life Safety Code, 1997.
NFPA 12, Carbon Dioxide Systems, 1993.
NFPA 13, Installation of Sprinkler Systems, 1996.
NFPA 14, Standpipe, Private Hydrant and Hose Systems, 1996.
NFPA 15, Water Spray Fixed Systems for Fire Protection, 1996.
NFPA 20, Installation of Stationary Pumps, 1996.
NFPA 2001, Clean Agent Fire Extinguishing Systems, 1996.
NFPA 22, Water Tanks for Private Fire Protection, 1996.
NFPA 221, Fire Walls and Fire Barrier Walls, 1997.
NFPA 24, Private Fire Service Mains and Their Appurtenances, 1995.
NFPA 25, Water Based Fire Protection Systems, 1995.
NFPA 30, Flammable and Combustible Liquids Code, 2003.
NFPA 5000, Building Construction and Safety Code, 2003.
NFPA 54, National Fuel Gas Code, 1996.
NFPA 55, Compressed & Liquefied Gases in Cylinders, 1993.
NFPA 58, Liquefied Petroleum Gas Code, 2001.
NFPA 600 Industrial Fire Brigades, 1996.
NFPA 70, National Electric Code, 1996.
NFPA 704, Standard System for the Identification of the Hazards of Materials for Emergency Response, 2001.
NFPA 72, National Fire Alarm Code, 1996.
NFPA 75, Electronic Computer/Data Processing Systems, 1995.
NFPA 780, Lightning Protection Systems, 1997.
NFPA 80, Fire Doors and Fire Windows, 1995.
NFPA 801, Fire Protection for Facilities Handling Radioactive Materials, 2003.
NFPA 80A, Exterior Fire Exposures, 1993.

Table 3.3-8 Utility and Support Systems Codes and Standards

NFPA 90A, Installation of Air Conditioning and Ventilating Systems, 1996.
NFPA 90B, Installation of Warm Air Heating and Air Conditioning Systems, 1996.
NFPA 91, Exhaust Systems for Air Conveying of Materials, 1995.
NFPA, Fire Protection Handbook, 18th Edition, Section 9, Chapter 30, Nuclear Facilities, 1997.
NFPA 110, Standard for Emergency and Standby Power Systems, 2002.
NFPA 111, Standard on Stored Electrical Energy Emergency and Standby Power Systems, 2001.
NFPA 70E, Standard for Electrical Safety Requirements for Employee Workplaces, 2000.
NFPA 79, Electrical Standard for Industrial Machinery, 1997.
PCI Design Handbook, Fifth Edition, 1999.
Uniform Building Code (UBC), 1997.
Uniform Mechanical Code (UMC), 1997.
Uniform Plumbing Code (UPC), 1997.

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4.0 RADIATION PROTECTION

This chapter describes the facility Radiation Protection Program. The Radiation Protection Program protects the radiological health and safety of workers and complies with the regulatory requirements in 10 CFR 19 (CFR, 2003a), 20 (CFR, 2003b) and 70 (CFR, 2003c).

This chapter includes radiation protection measures that are consistent with those previously submitted for Nuclear Regulatory Commission (NRC) review in Section 8 of the Louisiana Energy Services (LES) Claiborne Enrichment Center Safety Analysis Report (LES, 1993). These measures received regulatory approval in NUREG-1491, Safety Evaluation Report for the Claiborne Enrichment Center (NRC, 1994).

The information provided in this chapter, the corresponding regulatory requirement and the NRC acceptance criteria from NUREG-1520 (NRC, 2002), Chapter 4 are summarized in the table below. Information beyond that required by the Standard Review Plan is included. This additional information is an update of that previously submitted for the Claiborne Enrichment Center, as noted above.

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 4 Reference
Section 4.1 Commitment to Radiation Protection Program Implementation	10 CFR 20.1101, Subpart B	4.4.1.3
Section 4.2 Commitment to an ALARA Program	10 CFR 20.1101	4.4.2.3
Section 4.3 Organization and Personnel Qualifications	10 CFR 70.22	4.4.3.3
Section 4.4 Commitment to Written Procedures	10 CFR 70.22(8)	4.4.4.3
Section 4.5 Training Commitments	10 CFR 19.12 & 10 CFR 20.2110	4.4.5.3
Section 4.6 Ventilation and Respiratory Protection Programs Commitments	10 CFR 20, Subpart H	4.4.6.3
Section 4.7 Radiation Surveys and Monitoring Programs Commitments	10 CFR 20, Subparts F, C, L, M	4.4.7.3
Section 4.8 Contamination and Radiation Control	N/A	N/A
Section 4.9 Maintenance Areas - Methods and Procedures for Contamination Control	N/A	N/A
Section 4.10 Decontamination Policy and Provisions	N/A	N/A
Section 4.11 Additional Program Commitments	N/A	4.4.8.3

4.1 COMMITMENT TO RADIATION PROTECTION PROGRAM IMPLEMENTATION

The radiation program meets the requirements of 10 CFR 20 (CFR, 2003b), Subpart B, Radiation Protection Programs, and is consistent with the guidance provided in Regulatory Guide 8.2, Guide for Administrative Practice in Radiation Monitoring (NRC, 1973a). The facility develops, documents and implements its Radiation Protection Program commensurate with the risks posed by a uranium enrichment operation. The facility uses, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as reasonably achievable (ALARA). The radiation program content and implementation are reviewed at least annually as required by 10 CFR 20.1101(c) (CFR, 2003d). In addition, in accordance with 10 CFR 20.1101(d) (CFR, 2003d) constraints on atmospheric releases are established for the NEF such that no member of the public would be expected to receive a total effective dose equivalent in excess of 0.1 mSv/yr (10 mrem/yr) from these releases. Additional information regarding compliance with 10 CFR 20.1101(d) is provided in Section 9.2.

The facility's philosophy for radiation protection is reflected in the establishment of a Radiation Protection Program that has the specific purpose of maintaining occupational radiation exposures ALARA. This program includes written procedures, periodic assessments of work practices and internal/external doses received, work plans and the personnel and equipment required to help implement the ALARA goal.

The facility's administrative personnel exposure limits have been set below the limits specified in 10 CFR 20 (CFR, 2003b). This provides assurance that legal radiation exposure limits are not exceeded and that the ALARA principle is emphasized. The facility administrative exposure limits are given in Table 4.1-1, Administrative Radiation Exposure Limits. Estimates of the facility area radiation dose rates and individual personnel exposures, during normal operations, are shown in Table 4.1-2, Estimated Dose Rates and Table 4.1-3, Estimated Individual Exposures. These estimates are based upon the operating experience of similar Urenco facilities in Europe.

The annual dose equivalent accrued by a typical radiation worker at a uranium enrichment plant is usually low. At the Urenco Capenhurst plant, the maximum annual worker dose equivalent was 3.1 mSv (310 mrem), 2.2 mSv (220 mrem), 2.8 mSv (280 mrem), 2.7 mSv (270 mrem) and 2.3 mSv (230 mrem) during the years 1998 through 2002, respectively. For each of these same years, the average annual worker dose equivalent was approximately 0.2 mSv (20 mrem) (Urenco, 2000; Urenco, 2001; Urenco, 2002).

Protection of plant personnel requires (a) surveillance of and control over the radiation exposure of personnel; and (b) maintaining the exposure of all personnel not only within permissible limits, but "as low as is reasonably achievable," in compliance with applicable regulations and license conditions. The objectives of Radiation Protection are to prevent acute radiation injuries (nonstochastic or deterministic effects) and to limit the potential risks of probabilistic (stochastic) effects (which may result from chronic occupational exposure) to an acceptable level.

The radiation exposure policy and control measures for personnel are set up in accordance with requirements of 10 CFR 20 (CFR, 2003b) and the guidance of applicable Regulatory Guides. Recommendations from the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements (NCRP) may also be used in the formulation and evolution of the facility Radiation Protection Program.

The facility corrective action process is implemented if (1) personnel dose monitoring results or personnel contamination levels exceed the administrative personnel limits; or if an incident results in airborne occupational exposures exceeding the administrative limits or (2) the dose limits in 10 CFR 20 (CFR, 2003b), Appendix B or 10 CFR 70.61 (CFR, 2003e) are exceeded.

The information developed from the corrective action process is used to improve radiation protection practices and to preclude the recurrence of similar incidents. If an incident as described in item two above occurs, the NRC is informed of the corrective action taken or planned to prevent recurrence and the schedule established by the facility to achieve full compliance. The corrective action process and incident investigation process are described in Section 11.6, Incident Investigations and Corrective Action Process.

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to the general guidelines of the occupational radiation protection program and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994), Section 8.4.

4.1.1 Responsibilities of Key Program Personnel

In this section the Radiation Protection Program's organizational structure is described. The responsibilities of key personnel are also discussed. These personnel play an important role in the protection of workers, the environment and implementation of the ALARA program. Chapter 2, Organization and Administration, discusses the facility organization and administration in further detail. Section 2.2, Key Management Positions of Chapter 2, presents a detailed discussion of the responsibilities of key management personnel.

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to the responsibilities assigned to facility personnel and the extent of incorporation of the ALARA principle into the facility's radiation protection program and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994) Section 8.3.

4.1.1.1 Plant Manager

The Plant Manager is responsible for all aspects of facility operation, including the protection of all persons against radiation exposure resulting from facility operations and materials, and for compliance with applicable NRC regulations and the facility license.

4.1.1.2 Health, Safety and Environment Manager

The Health, Safety, and Environment (HS&E) Manager reports to the Plant Manager and has the responsibility for directing the activities that ensure the facility maintains compliance with appropriate rules, regulations, and codes. This includes HS&E activities associated with nuclear safety, radiation protection, chemical safety, environmental protection, and industrial safety. The HS&E Manager works with the other facility managers to ensure consistent interpretations of HS&E requirements, performs independent reviews and supports facility and operations change control reviews.

4.1.1.3 Radiation Protection Manager

The Radiation Protection Manager reports to the HS&E Manager. The Radiation Protection Manager is responsible for implementing the Radiation Protection Program. In matters involving radiological protection, the Radiation Protection Manager has direct access to the Plant Manager. The Radiation Protection Manager and his staff are responsible for:

- Establishing the Radiation Protection Program
- Generating and maintaining procedures associated with the program
- Assuring that ALARA is practiced by all personnel
- Reviewing and auditing the efficacy of the program in complying with NRC and other governmental regulations and applicable Regulatory Guides
- Modifying the program based upon experience and facility history
- Adequately staffing the Radiation Protection group to implement the Radiation Protection Program
- Establishing and maintaining an ALARA program
- Establishing and maintaining a respirator usage program
- Monitoring worker doses, both internal and external
- Complying with the radioactive materials possession limits for the facility
- Handling of radioactive wastes when disposal is needed
- Calibration and quality assurance of all radiological instrumentation, including verification of required Lower Limits of Detection or alarm levels
- Establishing and maintaining a radiation safety training program for personnel working in Restricted Areas

- Performing audits of the Radiation Protection Program on an annual basis
- Establishing and maintaining the radiological environmental monitoring program
- Posting the Restricted Areas, and within these areas, posting: Radiation, Airborne Radioactivity, High Radiation and Contaminated Areas as appropriate; and developing occupancy guidelines for these areas as needed.

4.1.1.4 Operations Manager

The Operations Manager is responsible for operating the facility safely and in accordance with procedures so that all effluents released to the environment and all exposures to the public and facility personnel meet the limits specified in applicable regulations, procedures and guidance documents.

4.1.1.5 Facility Personnel

Facility personnel are required to work safely and to follow the rules, regulations and procedures that have been established for their protection and the protection of the public. Personnel whose duties require (1) working with radioactive material, (2) entering radiation areas, (3) controlling facility operations that could affect effluent releases, or (4) directing the activities of others, are trained such that they understand and effectively carry out their responsibilities.

4.1.2 Staffing of the Radiation Protection Program

Only suitably trained radiation protection personnel are employed at the facility. For example, the Radiation Protection Manager has, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and three years of responsible nuclear experience associated with implementation of a Radiation Protection Program. At least two years of this nuclear experience is at a facility that processes uranium, including uranium in soluble form. Other members of the Radiation Protection Program staff are trained and qualified consistent with the guidance provided in American National Standards Institute (ANSI) standard 3.1, Selection, Qualification and Training of Personnel for Nuclear Power Plants (ANSI, 1993).

Sufficient resources in terms of staffing and equipment are provided to implement an effective Radiation Protection Program.

4.1.3 Independence of the Radiation Protection Program

The Radiation Protection Program remains independent of the facility's routine operations. This independence ensures that the Radiation Protection Program maintains its objectivity and is focused only on implementing sound radiation protection principles necessary to achieve occupational doses and doses to members of the public that are ALARA. It was previously noted in Section 4.1.1.3, Radiation Protection Manager, that in matters involving radiological protection, the Radiation Protection Manager has direct access to the Plant Manager.

4.1.4 Radiation Safety Committee

A Radiation Safety Committee meets periodically to review, in accordance with 10 CFR 20.1101(c) (CFR, 2003d), the status of projects, measure performance, look for trends and to review radiation safety aspects of facility operations. The Radiation Protection Manager chairs the Radiation Safety Committee. The other Radiation Safety Committee members come from quality assurance, operations, maintenance, and technical support, as deemed appropriate by the Plant Manager.

The objectives of the Radiation Safety Committee are to maintain a high standard of radiation protection in all facility operations. The Radiation Safety Committee reviews the content and implementation of the Radiation Protection Program at a working level and strives to improve the program by reviewing exposure trends, the results of audits, regulatory inspections, worker suggestions, survey results, exposure incidents, etc.

The maximum interval between meetings may not exceed 180 days. A written report of each Radiation Safety Committee meeting is forwarded to all Managers.

4.2 COMMITMENT TO AN ALARA PROGRAM

Section 4.1, Commitment to Radiation Protection Program Implementation, above states the facility's commitment to the implementation of an ALARA program. The objective of the program is to make every reasonable effort to maintain facility exposures to radiation as far below the dose limits of 10 CFR 20.1201 (CFR, 2003f) as is practical and to maintain radiation exposures to members of the public such that they are not expected to receive the dose limits of 10 CFR 20.1101(d) (CFR, 2003d). The design and implementation of the ALARA program is consistent with the guidance provided in Regulatory Guides 8.2 (NRC, 1973a), 8.13 (NRC, 1999a), 8.29 (NRC, 1996), and 8.37 (NRC, 1993g). The operation of the facility is consistent with the guidance provided in Regulatory Guide 8.10 (NRC, 1977).

Annual doses to individual personnel are maintained ALARA. In addition, the annual collective dose to personnel (i.e., the sum of all annual individual doses, expressed in person-Sv or person-rem) is maintained ALARA. The dose equivalent to the embyro/fetus is maintained below the limits of 10 CFR 20.1208 (CFR, 2003g).

The Radiation Protection Program is written and implemented to ensure that it is comprehensive and effective. The written program documents policies that are implemented to ensure the ALARA goal is met. Facility procedures are written so that they incorporate the ALARA philosophy into the routine operations of the facility and ensure that exposures are consistent with 10 CFR 20.1101 (CFR, 2003d) limits. As discussed in Section 4.7, Radiation Surveys and Monitoring Programs Commitments, radiological zones will be established within the facility. The establishment of these zones supports the ALARA commitment in that the zones minimize the spread of contamination and reduce unnecessary exposure of personnel to radiation.

Specific goals of the ALARA program include maintaining occupational exposures as well as environmental releases as far below regulatory limits as is reasonably achievable. The ALARA concept is also incorporated into the design of the facility. The size and number of areas with higher dose rates are minimized consistent with accessibility for performing necessary services in the areas. Areas where facility personnel spend significant amounts of time are designed to maintain the lowest dose rates reasonably achievable.

The Radiation Protection Manager is responsible for implementing the ALARA program and ensuring that adequate resources are committed to make the program effective. The Radiation Protection Manager prepares an annual ALARA program evaluation report. The report reviews (1) radiological exposure and effluent release data for trends, (2) audits and inspections, (3) use, maintenance and surveillance of equipment used for exposure and effluent control, and (4) other issues, as appropriate, that may influence the effectiveness of the radiation protection/ ALARA programs. Copies of the report are submitted to the Plant Manager and the Safety Review Committee.

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to the responsibilities assigned to facility personnel and the extent of incorporation of the ALARA principle in facility's radiation protection program and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994) Section 8.3.

4.2.1 ALARA Committee

The Safety Review Committee (SRC) fulfills the duties of the ALARA Committee. The SRC meets at least quarterly. Additional details concerning the membership and qualifications of the SRC are provided in Chapter 2, Organization and Administration.

Programs for improving the effectiveness of equipment used for effluent and exposure control are also evaluated by the SRC. The recommendations of the committee are documented in writing. The implementation of the committee's recommendations is tracked to completion via the Corrective Action Program, which is described in Section 11.6, Incident Investigations and Correction Action Process.

As part of its duties, the SRC reviews the effectiveness of the ALARA program and determines if exposures, releases and contamination levels are in accordance with the ALARA concept. It also evaluates the results of assessments made by the radiation protection organization, reports of facility radiation levels, contamination levels, and employee exposures for identified categories of workers and types of operations. The committee is responsible for ensuring that the occupational radiation exposure dose limits of 10 CFR 20 (CFR, 2003b) are not exceeded under normal operations. The committee determines if there are any upward trends in personnel exposures, environmental releases and facility contamination levels.

The ALARA program facilitates interaction between radiation protection and operations personnel. The SRC, comprising staff members responsible for radiation protection and operations, is particularly useful in achieving this goal. The SRC periodically reviews the goals and objectives of the ALARA program. The ALARA program goals and objectives are revised to incorporate, as appropriate, new technologies or approaches and operating procedures or changes that could cost-effectively reduce potential radiation exposures.

4.3 ORGANIZATION AND PERSONNEL QUALIFICATIONS

The regulation 10 CFR 70.22 (CFR, 2003h) requires that the technical qualifications, including training and experience of facility staff be provided in the license application. This information is provided in this section.

The Radiation Protection Program staff is assigned responsibility for implementation of the Radiation Protection Program functions. Only suitably trained radiation protection personnel are employed at the facility. Staffing is consistent with the guidance provided in Regulatory Guides 8.2 (NRC, 1973a) and 8.10 (NRC, 1977).

As previously discussed, the Radiation Protection Manager has, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and three years of responsible nuclear experience associated with implementation of a Radiation Protection Program. The nuclear experience includes at least two years of experience at a facility that processes uranium, including uranium in soluble form. As stated in Section 4.1.2, Staffing of the Radiation Protection Program, other members of the Radiation Protection Program staff are trained and qualified consistent with the guidance provided in American National Standards Institute (ANSI) standard 3.1, Selection, Qualification and Training of Personnel for Nuclear Power Plants (ANSI, 1993).

The Radiation Protection Manager reports to the HS&E Manager and has the responsibility for establishing and implementing the Radiation Protection Program. These duties include the training of personnel in use of equipment, control of radiation exposure of personnel, continuous determination and evaluation of the radiological status of the facility, and conducting the radiological environmental monitoring program. The facility organization chart establishes clear organizational relationships among the radiation protection staff and the other facility line managers. The facility operating organization is described in Chapter 2, Organization and Administration.

In all matters involving radiological protection, the Radiation Protection Manager has direct access to the Plant Manager. The Radiation Protection Manager is skilled in the interpretation of radiation protection data and regulations. The Radiation Protection Manager is also familiar with the operation of the facility and radiation protection concerns relevant to the facility. The Radiation Protection Manager is a resource for radiation safety management decisions.

4.4 COMMITMENT TO WRITTEN PROCEDURES

All operations at LES involving licensed materials are conducted through the use of procedures as required by 10 CFR 70.22(8) (CFR, 2003h). Radiation protection procedures are prepared, reviewed and approved to carry out activities related to the radiation protection program. Procedures are used to control radiation protection activities in order to ensure that the activities are carried out in a safe, effective and consistent manner. Radiation protection procedures are reviewed and revised as necessary, to incorporate any facility or operational changes or changes to the facility's Integrated Safety Analysis (ISA).

The radiation protection procedures are assigned to members of the radiation protection staff for development. Initial procedure drafts are reviewed by members of the facility staff, by personnel with enrichment plant operating experience, and other staff members as appropriate. The designated approver determines whether or not any additional, cross-disciplinary review is required. Changes to procedures are processed as follows. The writer documents the change as well as the reason for the change. The Radiation Protection Manager (or a designee who has the qualifications of the Radiation Protection Manager) reviews and approves procedures as well as proposed revisions to procedures. Final approval of the revised procedure is by the Plant Manager, or a designated alternate. Chapter 11, Management Measures, describes the program implemented for the control of procedures.

4.4.1 Radiation Work Permit Procedures

All work performed in Restricted Areas is performed in accordance with a Radiation Work Permit (RWP). The procedures controlling RWPs are consistent with the guidance provided in Regulatory Guide 8.10 (NRC, 1977). An RWP may also be required whenever the Radiation Protection Manager deems that one is necessary. Activities involving licensed materials not covered by operating procedures and where radioactivity levels are likely to exceed airborne radioactivity limits require the issuance of a RWP. Both routine and non-routine activities are performed under a RWP. The RWP provides a description of the work to be performed. That is, the RWP defines the authorized activities. The RWP summarizes the results of recent dose rate surveys, contamination surveys, airborne radioactivity results, etc. The RWP specifies the precautions to be taken by those performing the task. The specified precautions may include personal protective equipment to be worn while working (e.g., gloves, respirators, personnel monitoring devices), stay-times or dose limits for work in the area, record keeping requirements (e.g., time or dose spent on job) and the attendance of a radiation protection technician during the work. The RWP requires approval by the Radiation Protection Manager or designee. The designee must meet the requirements of Section 4.1.2, Staffing of the Radiation Protection Program. RWPs have a predetermined period of validity with a specified expiration or termination time.

Standing RWPs are issued for routinely performed activities, such as tours of the plant by shift personnel or the charging of cylinders. A Standing RWP would, for example, be used for the job evolution of cylinder charging; a new RWP is not issued each time a new cylinder is charged.

Listed below are requirements of the RWP procedures.

- The Radiation Protection Manager or designee is responsible for determining the need for, issuing and closing out RWPs
- Planned activities or changes to activities inside Restricted Areas or work with licensed materials are reviewed by the Radiation Protection Manager or designee for the potential to cause radiation exposures to exceed action levels or to produce radioactive contamination
- RWPs include requirements for any necessary safety controls, personnel monitoring devices, protective clothing, respiratory protective equipment, and air sampling equipment and the attendance of radiation protection technicians at the work location
- RWPs are posted at access points to Restricted Areas with copies of current RWPs posted at the work area location
- RWPs clearly define and limit the work activities to which they apply. A RWP is closed out when the applicable work activity for which it was written is completed and terminated
- RWPs are retained as a record at least for the life of the facility.

The subject matter discussed above is an improved version of the subject matter of Claiborne Enrichment Center SAR (LES, 1993). The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to the RWP system and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion on is in NUREG-1491 (NRC, 1994), Section 8.4.1.7.

4.5 TRAINING COMMITMENTS

The design and implementation of the radiation protection training program complies with the requirements of 10 CFR 19.12 (CFR, 2003i). Records are maintained in accordance with 10 CFR 20.2110 (CFR, 2003j).

The development and implementation of the radiation protection training program is consistent with the guidance provided in the following regulatory guidance documents:

- Regulatory Guide 8.10-Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable (NRC, 1977)
- Regulatory Guide 8.13-Instructions Concerning Prenatal Radiation Exposure (NRC, 1999a)
- Regulatory Guide 8.29-Instructions Concerning Risks From Occupational Radiation Exposure (NRC, 1996)
- ASTM C986-89-Developing Training Programs in the Nuclear Fuel Cycle (ASTM, 1989)
- ASTM E1168-95-Radiological Protection Training for Nuclear Facility Workers (ASTM, 1995).

All personnel and visitors entering Restricted Areas or Controlled Areas, as defined below, receive training that is commensurate with the radiological hazard to which they may be exposed. Alternatively, visitors will be provided with trained escorts who have received radiation protection training.

The level of radiation protection training is based on the potential radiological health risks associated with an employee's work responsibilities and incorporates the provisions of 10 CFR 19.12 (CFR, 2003i). In accordance with 10 CFR 19.12 (CFR, 2003i), any individual working at the facility who is likely to receive in a year a dose in excess of 1 mSv (100 mrem) is:

A. Kept informed of the storage, transfer, or use of radioactive material

B. Instructed in the health protection problems associated with exposure to radiation and radioactive material, in precautions or procedures to minimize exposure, and in the purposes and functions of protective devices employed

C. Required to observe, to the extent within the worker's control, the applicable provisions of the NRC regulations and licenses for the protection of personnel from exposure to radiation and radioactive material

D. Instructed of their responsibility to report promptly to the facility management, any condition which may cause a violation of NRC regulations and licenses or unnecessary exposure to radiation and radioactive material

E. Instructed in the appropriate response to warnings made in the event of any unusual occurrence or malfunction that may involve exposure to radiation and radioactive material

F. Advised of the various notifications and reports to individuals that a worker may request in accordance with 10 CFR 19.13 (CFR, 2003k).

The radiation protection training program takes into consideration a worker's normally assigned work activities. Abnormal situations involving exposure to radiation and radioactive material, which can reasonably be expected to occur during the life of the facility, are also evaluated and factored into the training. The extent of these instructions is commensurate with the potential radiological health protection problems present in the work place.

Retraining of personnel previously trained is performed for radiological, chemical, industrial, and criticality safety at least annually. The retraining program also includes procedure changes, and updating and changes in required skills. Changes to training are implemented, when required, due to incidents potentially compromising safety or if changes are made to the facility or processes. Records of training are maintained in accordance with LES records management system. Training programs are established in accordance with Section 11.3, Training and Qualifications. The radiation protection sections of the training program are evaluated at least annually. The program content is reviewed to ensure it remains current and adequate to assure worker safety.

The specifics of the Radiation Protection Training are described in the following section.

4.5.1 Radiation Protection Training

Radiation protection training is highlighted to emphasize the high level of importance placed on the radiological safety of plant personnel and the public. In-depth radiation protection training is provided for the various types of job functions (e.g., production operator, radiation protection technician, contractor personnel) commensurate with the radiation safety responsibilities associated with each such position. Visitors to a Restricted Area are trained in the formal training program or are escorted by trained personnel while in the Restricted Area.

Personnel access procedures ensure the completion of formal nuclear safety training prior to permitting unescorted access into the Restricted Areas. Training sessions covering criticality safety, radiation protection and emergency procedures are conducted on a regular basis to accommodate new employees or those requiring retraining. Retraining is conducted when necessary to address changes in policies, procedures, requirements and the ISA.

Specific topics covered in the training program are listed in Chapter 11, Management Measures, Section 11.3.3.1.1. The training provided includes the requirements of 10 CFR 19 (CFR, 2003a).

Individuals attending these sessions must pass an initial examination covering the training contents to assure the understanding and effectiveness of the training. The effectiveness and adequacy of the training program curriculum and instructors are also evaluated by audits performed by operational area personnel responsible for criticality safety and radiation protection.

Since contractor employees may perform diverse tasks in the Restricted Areas or Controlled Areas of the facility, formal training for these employees is designed to address the type of work they perform. In addition to applicable radiation safety topics, training contents may include RWPs, special bioassay sampling, and special precautions for welding, cutting, and grinding. Instructors certified by the Radiation Protection Manager conduct the radiation protection training programs.

The Radiation Protection Manager is responsible for establishing and maintaining the radiation protection training for all personnel, including contractor personnel who may be working at the facility. Records are maintained for each employee documenting the training date, scope of the training, identity of the trainer(s), any test results and other associated information.

Individuals requiring unescorted access to a Restricted Area receive annual retraining. Contents of the formal radiation protection training program are reviewed and updated as required at least every two years by the HS&E Manager and Radiation Protection Manager to ensure that the programs are current and adequate.

4.6 VENTILATION AND RESPIRATORY PROTECTION PROGRAMS COMMITMENTS

The regulations contained in 10 CFR 20 (CFR, 2003b), Subpart H, define the required elements of the facility respiratory protection and ventilation programs. This section describes the design and management measures taken to ensure that the installed ventilation and containment systems operate effectively. This section also describes the worker respiratory protection program.

The design of the ventilation and respiratory protection programs is consistent with the guidance contained in the following documents:

- Regulatory Guide 8.24-Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication (NRC, 1979)
- ANSI N510-1980-Testing of Nuclear Air Cleaning Systems (ANSI,1980)
- ERDA 76-21-Nuclear Air Cleaning Handbook (ERDA,1976)
- NCRP Report No. 59-Operational Radiation Safety Program (NCRP,1978)
- Regulatory Guide 8.15-Acceptable Programs for Respiratory Protection (NRC,1999b)
- ANSI Z88.2-1992-Practices for Respiratory Protection (ANSI,1992).

4.6.1 Ventilation Program

The confinement of uranium and the attenuation of its associated radiation are a design requirement for the facility. The internal radiation exposure of workers is controlled primarily by the containment of UF_6 within process equipment. The entire UF_6 enrichment process, except for liquid sampling, is operated under a partial vacuum so that leaks are into the system and not into work areas.

Ventilation systems for the various buildings control the temperature and the humidity of the air inside the building. The ventilation systems serving normally non-contaminated areas exhaust approximately 10% of the air handled to the atmosphere. Ventilation systems serving potentially contaminated areas include design features that provide for confinement of radiological contamination. Ventilation systems for potentially contaminated areas exhaust 100% of the air handled to the environment through the exhaust stacks. All air released from potentially contaminated areas is filtered to remove radioactive particulates before it is released. The ventilation systems for potentially contaminated areas are designed to maintain the potentially contaminated areas at a slightly negative pressure relative to the uncontaminated areas. This ensures that the airflow direction is from areas of little or no contamination to areas of higher contamination.

Process vents from the Separations Building Module are collected by the Separations Building Gaseous Effluent Vent System (GEVS). Some areas of the Technical Services Building (TSB) also have fume hoods that are connected to the TSB GEVS. Air released from the Centrifuge Test Facility and the Centrifuge Post Mortem Facilities is filtered by the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System prior to release. The systems operate slightly below atmospheric pressure to remove potentially hazardous vapors and particulate from confined areas of the plant. The systems contain particulate and carbon adsorption filters to remove radioactive materials from the gas stream prior to release from the plant. Continuous HF monitors are provided upstream of the filters with high level alarms to inform operators of UF_6 releases in the plant.

Normal operation of the facility will not result in a release of radioactive material that exceeds regulatory limits. Ventilation systems for areas that do not have the potential for contamination are not monitored for radioactivity because radioactive material is not handled or processed in these areas. No emergency ventilation systems are provided for operation when the normal ventilation systems are shut down.

Several measures are in place to ensure effective operation of the ventilation systems. Differential pressure across High Efficiency Particulate Air (HEPA) filters in potentially contaminated ventilation exhaust systems is monitored monthly or automatically monitored and alarmed. Operating procedures specify limits and set points on the differential pressure consistent with manufacturers' recommendations. Filters are changed if they fail to function properly or if the differential pressure exceeds the manufacturers' ratings.

Filter inspection, testing, maintenance and change out criteria are specified in written procedures approved by the Technical Services Manager, or a designated alternate. Change-out frequency is based on considerations of filter loading, operating experience, differential pressure data and any UF_6 releases indicated by HF alarms.

Gloveboxes are designed to maintain a negative differential pressure of about 0.623 mbar (0.25 in H_2O). This differential pressure is maintained anytime that the glovebox is in use. If the differential pressure is lost, use of the glovebox is suspended until the required differential pressure is restored.

Air flow rates at exhausted enclosures and close-capture points, when in use, are adequate to preclude escape of airborne uranium and minimize the potential for intake by workers. Air flow rates are checked monthly when in use and after modification of any hood, exhausted enclosure, close-capture point equipment or ventilation system serving these barriers.

The various programs that pertain to preventive and corrective maintenance are described in Chapter 11, Sections 11.2.2, Corrective Maintenance and 11.2.3, Preventive Maintenance respectively.

4.6.2 Respiratory Protection Program

The facility uses process and engineering controls to control the concentration of radioactive material in air. However, there may be instances when it is not practical to apply process or other engineering controls. When it is not possible to control the concentrations of radioactive material in the air to values below those that define an airborne radioactivity area, other means are implemented to maintain the total effective dose equivalent ALARA. In these cases, the ALARA goal is met by an increase in monitoring and the limitation of intakes by one or more of the following means:

A. Control of access

B. Limitation of exposure times

C. Use of respiratory protection equipment

D. Other controls, as available and appropriate.

If an ALARA analysis is performed to determine whether or not respirators should be used, safety factors other than radiological factors may be considered. The impact of respirator use on workers' industrial health and safety is factored into decisions to use respirators.

If the decision is made to permit the use of respiratory protection equipment to limit the intake of radioactive material, only National Institute of Occupational Safety and Health (NIOSH) certified equipment is used. The respiratory protection program meets the requirements of 10 CFR 20 (CFR, 2003b), Subpart H (Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas).

The respiratory protection program includes the following elements:

A. Air sampling to identify the potential hazard, select proper equipment and estimate doses

B. Surveys and, when necessary, bioassays to evaluate actual intakes

C. Performance testing of respirators for operability (user seal check for face sealing devices and functional check for others) immediately prior to each use.

D. Written procedures for the following:

    1. Monitoring, including air sampling and bioassays

    2. Supervision and training of respirator users

    3. Fit testing

    4. Respirator selection

5. Breathing air quality

6. Inventory and control

7. Storage, issuance, maintenance, repair, testing, and quality assurance of respiratory protection equipment

8. Record keeping

9. Limitations on periods of respirator use and relief from respirator use.

E. Determination by a physician that the individual user is medically fit to use respiratory protection equipment:

1. Before the initial fitting of a face sealing respirator

2. Before the first field use of non-face sealing respirators

3. Either every 12 months thereafter, or periodically at a frequency determined by a physician.

F. A respirator fit test requires a minimum fit factor of at least 10 times the Assigned Protection Factor (APF) for negative pressure devices, and a fit factor of at least 500 times the APF for any positive pressure, continuous flow, and pressure-demand devices. The fit testing is performed before the first field use of tight fitting, face-sealing respirators. Subsequent testing is performed at least annually thereafter. Fit testing must be performed with the facepiece operating in the negative pressure mode.

1. Each user is informed that they may leave the area at any time for relief from respirator use in the event of equipment malfunction, physical or psychological distress, procedural or communication failure, significant deterioration of operating conditions, or any other conditions that might require such relief.

2. In the selection and use of respirators, the facility provides for vision correction, adequate communication, low temperature work environments, and the concurrent use of other safety or radiological protection equipment. Radiological protection equipment is used in such a way as not to interfere with the proper operation of the respirator.

3. Standby rescue persons are used whenever one-piece atmosphere-supplying suits are in use. Standby rescue personnel are also used when any combination of supplied air respiratory protection device and personnel protective equipment is in use that presents difficulty for the wearer to remove the equipment. The standby personnel are equipped with respiratory protection devices or other apparatus appropriate for the potential hazards. The standby rescue personnel observe and maintain continuous communication with the workers (visual, voice, signal line, telephone, radio, or other suitable means). The rescue personnel are immediately available to assist the workers in case of a failure of the air supply or

for any other emergency. The Radiation Protection Manager specifies the number of standby rescue personnel that must be immediately available to assist all users of this type of equipment and to provide effective emergency rescue if needed.

4. Atmosphere-supplying respirators are supplied with respirable air of grade D quality or better as defined by the Compressed Gas Association in publication G-7.1, Commodity Specification for Air, (CGA, 1997) and included in the regulations of the Occupational Safety and Health Administration (29 CFR 1910.134(i)(1)(ii)(A) through (E) (CFR, 2003l)).

5. No objects, materials or substances (such as facial hair), or any conditions that interfere with the face-to-facepiece seal or valve function, and that are under the control of the respirator wearer, are allowed between the skin of the wearer's face and the sealing surface of a tight-fitting respirator facepiece.

The dose to individuals from the intake of airborne radioactive material is estimated by dividing the ambient air concentration outside the respirator by the assigned protection factor. If the actual dose is later found to be greater than that estimated initially, the corrected value is used. If the dose is later found to be less than the estimated dose, the lower corrected value may be used.

Records of the respiratory protection program (including training for respirator use and maintenance) are maintained in accordance with the facility records management program as described in Section 11.7, Records Management. Respiratory protection procedures are revised as necessary whenever changes are made to the facility, processing or equipment.

4.7 RADIATION SURVEYS AND MONITORING PROGRAMS COMMITMENTS

Radiation surveys are conducted for two purposes: (1) to ascertain radiation levels, concentrations of radioactive materials, and potential radiological hazards that could be present in the facility; and (2) to detect releases of radioactive material from facility equipment and operations. Radiation surveys will focus on those areas of the facility identified in the ISA where the occupational radiation dose limits could potentially be exceeded. Measurements of airborne radioactive material and/or bioassays are used to determine that internal occupational exposures to radiation do not exceed the dose limits specified in 10 CFR 20 (CFR, 2003b), Subpart C.

To assure compliance with the requirements of 10 CFR 20 (CFR, 2003b) Subpart F, there are written procedures for the radiation survey and monitoring programs. The radiation survey and monitoring programs assure compliance with the requirements of 10 CFR 20 (CFR, 2003b) Subpart F (Surveys and Monitoring), Subpart C (Occupational Dose Limits), Subpart L (Records) and Subpart M (Reports).

The radiation survey and monitoring programs are consistent with the guidance provided in the following references:

- Regulatory Guide 8.2-Guide for Administrative Practice in Radiation Monitoring (NRC,1973a)
- Regulatory Guide 8.4-Direct-Reading and Indirect-Reading Pocket Dosimeters (NRC,1973b)
- Regulatory Guide 8.7- Instructions for Recording and Reporting Occupational Radiation Exposure Data (NRC, 1992a)
- Regulatory Guide 8.9-Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program (NRC,1993f)
- Regulatory Guide 8.24-Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication (NRC,1979)
- Regulatory Guide 8.25-Air Sampling in the Workplace (NRC, 1992b)
- Regulatory Guide 8.34-Monitoring Criteria and Methods To Calculate Occupational Radiation Doses (NRC, 1992c)
- NUREG-1400-Air Sampling in the Workplace (NRC,1993a)
- ANSI/HPS N13.1-1999-Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities (ANSI, 1999)
- ANSI N323-1978-Radiation Protection Instrumentation Test and Calibration (ANSI,1978)

- ANSI N13.11-1983-Dosimetry-Personnel Dosimetry Performance-Criteria for Testing (ANSI, 1983)

- ANSI N13.15-1985-Radiation Detectors-Personnel Thermoluminescence Dosimetry Systems-Performance (ANSI,1985)

- ANSI/HPS N13.22-1995-Bioassay Program for Uranium (ANSI,1995)

- ANSI N13.27-1981-Performance Requirements for Pocket-Sized Alarm Dosimeters and Alarm Ratemeters (ANSI,1981)

- ANSI/HPS N13.30-1996-Performance Criteria for Radiobioassay (ANSI,1996)

- ANSI N13.6-1966 (R1989), Practice for Occupational Radiation Exposure Records Systems (ANSI,1989)

The procedures include an outline of the program objectives, sampling procedures and data analysis methods. Equipment selection is based on the type of radiation being monitored. Procedures are prepared for each of the instruments used and specify the frequency and method of calibration. Maintenance and calibration are in accordance with the manufacturers' recommendations. Specific types of instruments used in the facility are discussed below.

The survey program procedures also specify the frequency of measurements and record keeping and reporting requirements. As stated in Section 4.1, Commitment to Radiation Protection Program Implementation, the facility corrective action process is implemented if: 1) personnel dose monitoring results or personnel contamination levels exceed the administrative personnel limits; or if an incident results in airborne occupational exposures exceeding the administrative limits, or 2) the dose limits in 10 CFR 20, Appendix B (CFR, 2003m) or 10 CFR 70.61 (CFR, 2003e) are exceeded. In the event the occupational dose limits given in 10 CFR 20 (CFR, 2003b), Subpart C are exceeded, notification of the NRC is in accordance with the requirements of 10 CFR 20, Subpart M—Reports.

All personnel who enter Restricted Areas (as defined below) are required to wear personnel monitoring devices that are supplied by a vendor that holds dosimetry accreditation from the National Voluntary Laboratory Accreditation Program. In addition, personnel are required to monitor themselves prior to exiting Restricted Areas which may have the potential for contamination.

Continuous airborne radioactivity monitors provide indication of the airborne activity levels in the Restricted Areas of the facility. Monitoring instruments for airborne alpha emitters are provided at different locations throughout facility. These monitors are designed to detect alpha emitters in the air, which would indicate the potential for uranium contamination. When deemed necessary, portable air samplers may be used to collect a sample on filter paper for subsequent analysis in the laboratory.

Monitor data is collected for regular analysis and documentation. Monitors in locations classified as Airborne Radioactivity Areas are equipped with alarms. The alarm is activated when airborne radioactivity levels exceed predetermined limits. The limits are set with consideration being given to both toxicity and radioactivity. The volume of air sampled may have to be adjusted to ensure adequate sensitivity with minimum sampling time. The operating history of the facility, changes in technology, changes in room functions and design, and changes in regulations may necessitate adjustment of the monitors.

Continuous monitoring of direct radiation exposure rates is not performed because the uranium processed in the facility is handled in closed containers. The radionuclides of interest are primarily alpha and beta emitters. The decay data and decay chains for these radionuclides are shown in Table 4.7-1, Radiation Emitted from Natural UF_6 Feed, and Figure 4.7-1, Uranium and Decay Products of Interest, respectively.

Alpha and beta radiation cannot penetrate the container walls. Typical area radiation monitors measure gamma radiation. At this facility, the gamma radiation is not present at sufficient levels to provide representative indications. Instead, periodic radiation monitoring is performed with portable survey meters and "wipe tests" for contamination are taken to evaluate radiological conditions in the facility.

A calibration is performed in accordance with written established procedures and documented prior to the initial use of each airflow measurement instrument (used to measure flow rates for air or effluent sampling) and each radioactivity measurement instrument. Periodic operability checks are performed in accordance with written established procedures. Calibrations are performed and documented on each airflow measurement and radioactivity measurement instrument at least annually (or according to manufacturers' recommendations, whichever is more frequent) or after failing an operability check, or after modifications or repairs to the instrument that could affect its proper response, or when it is believed that the instrument has been damaged.

Unreliable instruments are removed from service until repairs are completed. Portal monitors, hand and foot monitors and friskers have the required sensitivity to detect alpha contamination on personnel to ensure that radioactive materials do not spread to the areas outside the Restricted Areas. Instruments are calibrated with sources that are within ±5% of the reference value and are traceable to the National Institute of Standards and Technology or equivalent.

The background and efficiency of laboratory counting instruments, when used for radiation protection purposes, is determined daily. This determination may be less frequent only if necessary due to long counting intervals.

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to the instrument, calibration and maintenance program and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994), Section 8.4.1.6.

4.7.1 Radiological Zones

Radiological zones within the facility have been established to (1) control the spread of contamination, (2) control personnel access to avoid unnecessary exposure of personnel to radiation, and (3) control access to radioactive sources present in the facility. Table 4.1-2, Estimated Dose Rates, lists general dose rate estimates for the facility. These dose estimates were prepared based upon historical data from operating Urenco centrifuge enrichment facilities. Areas associated with higher dose rates may be restricted from public access, as determined by facility management. Areas where facility personnel spend substantial amounts of time are designed to minimize the exposure received when routine tasks are performed, in accordance with the ALARA principle.

The following definitions of areas are provided to describe how the facility Radiation Protection Program is implemented to protect workers and the general public on the site.

4.7.1.1 Unrestricted Area

NRC regulation 10 CFR 20.1003 (CFR, 2003n) defines an Unrestricted Area as an area, access to which is neither limited nor controlled by the licensee. The area adjacent to the facility site where LES does not normally exercise access control is an Unrestricted Area. This area can be accessed by members of the public, indigenous wildlife, or by facility personnel. The Unrestricted Area is governed by the limits in 10 CFR 20.1301 (CFR, 2003o). The total effective dose equivalent to individual members of the public from the licensed operation may not exceed 1 mSv (100 mrem) in a year (exclusive of background radiation). The dose in any Unrestricted Area from external sources may not exceed 0.02 mSv (2 mrem) in any one hour. In addition to the NRC limit, the Environmental Protection Agency, in 40 CFR 190 (CFR, 2003p), imposes annual dose equivalent limits of 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid, and 0.25 mSv (25 mrem) to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials to the general environment from uranium fuel cycle operations and to radiation from these operations.

4.7.1.2 Restricted Area

The NRC defines a Restricted Area as an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Access to and egress from a Restricted Area at the plant site is through a radiation protection control point known as a Monitor Station. Monitoring equipment is located at these egress points. All personnel are required to monitor themselves prior to exiting Restricted Areas that have the potential for contamination, using monitoring instruments that detect gross alpha contamination.

Examples of Restricted Areas include storage areas for UF_6 in the Cylinder Receipt and Dispatch Building and the potentially contaminated areas in the Technical Services Building. Personnel who have not been trained in radiation protection procedures are not allowed to access a Restricted Area without escort by trained personnel.

The areas defined below may exist within a Restricted Area. These areas may be temporary or permanent. The areas are posted to inform workers of the potential hazard in the area and to help prevent the spread of contamination. These areas are conspicuously posted in accordance with the requirements of 10 CFR 20.1902 (CFR, 2003q).

- An area in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.05 mSv (5 mrem) in 1 hr at 30 cm (11.8 in) from the radiation source or from any surface that the radiation penetrates is designated a "Radiation Area" as defined in 10 CFR 20.1003 (CFR, 2003n).

- An "Airborne Radioactivity Area" means a room, enclosure, or area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations (1) In excess of the derived air concentrations (DACs) specified in Appendix B (CFR, 2003m), to 10 CFR 20.1001 - 20.2401, or (2) To such a degree that an individual present in the area without respiratory protective equipment could exceed, during the hours an individual is present in a week, an intake of 0.6% of the annual limit on intake (ALI) or 12 DAC-hours. Note that entry into this area does not automatically require the wearing of a respirator.

- A "High Radiation Area" is an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 1 mSv (100 mrem) in 1 hour at 30 cm (11.8 in) from the radiation source or from any surface that the radiation penetrates. No examples of this type of area are expected during routine operation of the facility. This designation is provided here only for the purposes of emergency situations (drills and actual events).

- LES defines a "Contaminated Area" as an area where removable contamination levels are above 0.33 Bq/100 cm^2 (20 dpm/100 cm^2) of alpha activity or 16.7 Bq/100 cm^2 (1,000 dpm/100 cm^2) beta/gamma activity.

The NRC limits the soluble uranium intake of an individual to 10 milligrams in a week in consideration of chemical toxicity. LES posts areas where the intake of soluble uranium in one week is likely to exceed 1 milligram, if respiratory protection is not utilized.

4.7.1.3 Controlled Area

The NRC defines a Controlled Area as an area, outside of a Restricted Area but inside the site boundary, access to which can be limited by the licensee for any reason. The area of the plant within the perimeter fence but outside any Restricted Area is part of the Controlled Area. Due to the presence of the fence, members of the public do not have direct access to this Controlled Area of the site and must be processed by security and authorized to enter the site. Training for access to a Controlled Area is provided commensurate with the radiological hazard.

Site visitors include delivery people, tour guests and service personnel who are temporary, transient occupants of the Controlled Area. Area monitoring demonstrates compliance with public exposure limits for such visitors. All individuals who are contractor or LES employees and who work only in the Controlled Area are subject to the exposure limits for members of the public (CFR, 2003b).

4.7.2 Access and Egress Control

The facility establishes and implements an access control program that ensures that (a) signs, labels, and other access controls are properly posted and operative, (b) restricted areas are established to prevent the spread of contamination and are identified with appropriate signs, and (c) step-off pads, change facilities, protective clothing facilities, and personnel monitoring instruments are provided in sufficient quantities and locations.

Because there are no High Radiation Areas in the facility, there are no areas where access is physically prevented due to radiation level. Access control is by administrative methods. Access to certain areas may be physically prevented for security reasons. Personnel who have not been trained in radiation protection procedures are not allowed access to a Restricted Area without escort by other trained personnel.

Access to and egress from a Restricted Area is through one of the monitor stations at the particular Restricted Area boundary. Access to and egress from each Radiation Area, High Radiation Area, Contaminated Area or Airborne Radioactivity Area within a Restricted Area may also be individually controlled. A monitor (frisker), step-off pad and container for any discarded protective clothing may be provided at the egress point from certain of these areas to prevent the spread of contamination.

Action levels for skin and personal clothing contamination at the point of egress from Restricted Areas and any additional designated areas within the Restricted Area (e.g., a Contaminated Area which is provided with a step-off pad and frisker) shall not exceed 2.5 Bq/100 cm^2 (150 dpm/100 cm^2) alpha or beta/gamma contamination (corrected for background). Clothing contaminated above egress limits shall not be released unless it can be laundered to within these limits. If skin or other parts of the body are contaminated above egress limits, reasonable steps that exclude abrasion or other damage shall be undertaken to effect decontamination.

4.7.3 Posting for Radiation Protection Awareness

Restricted Areas and other areas within the Restricted Areas (e.g., Airborne Radioactivity Area) are clearly identified by physical means such as placarding or boundary marking, so that facility personnel can identify these areas and use their training to minimize their exposure. This identification is done in accordance with 10 CFR 20.1902 (CFR, 2003q). The radiation and contamination levels from the most recent survey are clearly noted on each posting.

4.7.4 Protective Clothing and Equipment

The proper use of protective clothing and equipment can minimize internal and external exposures to radioactivity. Personnel working in areas that are classified as Airborne Radioactivity Areas or Contaminated Areas must wear appropriate protective clothing. If the areas containing the surface contamination can be isolated from adjacent work areas via a barrier such that dispersible material is not likely to be transferred beyond the area of contamination, personnel working in the adjacent area are not required to wear protective clothing. Areas requiring protective clothing are posted at each of their entry points.

Radiation protection management and associated technical staff are responsible for determining the need for protective clothing in each work area. Areas requiring protective clothing are identified by posting signs at all area entry points.

4.7.5 Personnel Monitoring for External Exposures

External exposures are received primarily from the radioactive decay products of ^{235}U and ^{238}U. Most notably these progeny are ^{231}Th (several gammas, all low energy and low abundance), ^{234}Th (several gammas, most low abundance and low energy), and ^{234}Pa and ^{234m}Pa (many gammas, variable abundance, low and high energy). The ^{234m}Pa is the primary gamma source and is expected to contribute to a significant portion of the external exposure. Over the life of the facility, the number of tails-containing Uranium Byproduct Cylinders (UBCs) placed on the storage pad may increase to the pad's design capacity. In addition, the CRDB may reach its design capacity of feed and product cylinders. As a result, it is possible that the neutron contribution to the total worker dose may require monitoring. The neutrons are due to spontaneous fission in uranium as well as the alpha, neutron reaction on fluorine. Workers receive training regarding ALARA concepts such as time-distance-shielding to minimize their exposures.

All personnel whose duties require them to enter Restricted Areas wear individual external dosimetry devices, e.g., thermoluminescent dosimeters (TLDs) that are sensitive to beta, gamma and neutron radiation. Appropriate neutron survey meters are also available to the Radiation Protection staff. External dosimetry devices are evaluated at least quarterly to ascertain external exposures. Administrative limits on radiation exposure are provided in Table 4.1-1, Administrative Radiation Exposure Limits.

If 25% of the annual administrative limit (i.e., 2.5 mSv or 250 mrem) is exceeded in any quarter, then an investigation is performed and documented to determine what types of activities may have contributed to the worker's external exposure. The administrative limit already reflects ALARA principles, so this action level is appropriate. This investigation may include, but is not limited to procedural reviews, efficiency studies of the air handling system, cylinder storage protocol, and work practices.

Anytime an administrative limit is exceeded, the Radiation Protection Manager is informed. The Radiation Protection Manager is responsible for determining the need for and recommending investigations or corrective actions to the responsible Manager(s). Copies of the Radiation Protection Manager's recommendations are provided to the Safety Review Committee.

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to administrative radiation exposure limits and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994), Section 8.4.1.1.

4.7.6 Personnel Monitoring for Internal Exposures

Internal exposures for all personnel wearing external dosimetry devices are evaluated via direct bioassay (e.g. *in vivo* body counting), indirect bioassay (e.g., urinalysis), or an equivalent technique. For soluble (Class D) uranium, 10 CFR 20.1201(e) (CFR, 2003f) limits worker intake to no more than 10 milligrams of soluble uranium in a week. This is to protect workers from the toxic chemical effects of inhaling Class D uranium. The facility annual administrative limit for the Total Effective Dose Equivalent (TEDE) is 10 mSv (1000 mrem). Internal doses are evaluated at least annually.

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to proposed intake limits on soluble uranium and the 10 mSv (1000 mrem) TEDE and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994), Section 8.4.1.

Continuous air monitoring in Airborne Radioactivity Areas may be performed to complement the bioassay program. Alarm setpoints on the continuous air monitors in the Airborne Radioactivity Areas may be used to provide an indication that internal exposures may be approaching the action limit.

If the facility annual administrative limit is exceeded as determined from bioassay results, then an investigation is performed and documented to determine what types of activities may have contributed to the worker's internal exposure. The action limit is based on ALARA principles. Other factors such as the biological elimination of uranium are considered. This investigation may include, but is not limited to procedural reviews, efficiency studies of the air handling system, and work practices.

4.7.7 Evaluation of Doses

Dose evaluations may be performed at more frequent intervals and should be performed when reasonable suspicion exists regarding an abnormal exposure. The internal and external exposure values are summed in accordance with 10 CFR 20.1202 (CFR, 2003r). Procedures for the evaluation and summation of doses are based on the guidance contained in Regulatory Guides 8.7 (NRC, 1992a) and 8.34 (NRC, 1992c).

4.7.8 Monitor Stations

Monitor stations are the entry and exit points for Restricted Areas. Monitors are provided to detect radioactive contamination on personnel and their personal items, including hard hats. All personnel are required to monitor themselves, any hand-carried personal items, and hard hats prior to exiting a Restricted Area. Radiation protection management is responsible for Monitor Station provision and maintenance. Figure 4.7-2, Projected Radiological Zones shows the anticipated Restricted Areas. Monitor Station locations are evaluated and moved as necessary in response to changes in the facility radiological conditions.

4.7.9 Locker Rooms

Locker rooms for men and women are provided for personnel to change into appropriate work clothing and store personal belongings. The following facilities are provided for in the locker room area:

- Shower Rooms - shower rooms for men and women are provided as a place for personnel to wash/clean up after work. These shower rooms are not intended for personnel decontamination.

- Restrooms - restrooms for men and women are provided. These rooms are not for personnel decontamination.

- First Aid Station - a first aid station is provided to treat injured personnel.

- Personnel Decontamination Area - a personnel decontamination area is provided to handle cases of accidental radioactive contamination. A handwashing sink and a shower are provided for contamination removal.

- Information Area - an information area is provided to notify personnel of information important to radiation protection.

4.7.10 Storage Areas

Storage areas are provided for the following items:

- Protective (i.e., anti-contamination) clothing

- Respiratory protection equipment

- Shower rooms supplies

- Radiation protection supplies.

4.8 CONTAMINATION AND RADIATION CONTROL

The goal of maintaining occupational internal and external radiation exposures ALARA encompasses the individual's dose as well as the collective dose of the entire working population. Since the total effective dose equivalent (TEDE) is the sum of the internal and external exposures, the Radiation Protection Program addresses both contamination control and external radiation protection.

Listed below are examples of design and operating considerations that are implemented at the facility to reduce personnel radiation exposures:

- The enrichment process, with the exception of the Liquid Sampling part, is maintained under sub atmospheric pressure. The constant containment of UF_6 precludes direct contact with radioactive materials by personnel.
- Self-monitoring is required upon exit from Restricted Areas. Personnel are required to notify a member of the radiation protection staff if contamination is detected.
- All personnel are trained in emergency evacuation procedures in accordance with the facility Emergency Plan.
- Air flow rates at exhausted enclosures and close-capture points, when in use, are adequate to preclude escape of airborne uranium and minimize the potential for intake by workers. Air flow rates are checked monthly when in use and after modification of any hood, exhausted enclosure, close-capture point equipment or ventilation system serving these barriers.

4.8.1 Internal Exposures

Because the radionuclides present in this facility under routine operations are primarily alpha and beta emitters (with some low-energy gamma rays), the potential for significant internal exposure is greater than that for external exposure. Parameters important to determining internal doses are:

- The quantity of radioactive material taken into the body
- The chemical form of the radioactive material
- The type and half-life of radionuclide involved
- The time interval over which the material remains in the body.

The principal modes by which radioactive material can be taken into the body are:

- Inhalation
- Ingestion

- Absorption through the skin
- Injection through wounds.

4.8.1.1 Bioassay

Internal radiological exposures are evaluated annually as noted in Section 4.7.7, Evaluation of Doses. Based on the results of air sample monitoring data, bioassays are performed for all personnel who are likely to have had an intake of one milligram of uranium during a week. This is 10% of the 10 mg (3.5 E-4 oz) in a week regulatory limit (10 CFR 20.1201(e) (CFR, 2003f)) for intake of Class D uranium. The bioassay program has a sensitivity of 5 μg/L (7 E-7 oz/gal) of uranium concentration, assuming that the sample is taken within ten days of the postulated intake and that at least 1.4 L (0.37 gal) of sample is available from a 24-hour sampling period. Until urinalysis results indicate less than 15 μg/L (2.0 E-6 oz/gal) of uranium concentration, workers are restricted from activities that could routinely or accidentally result in internal exposures to soluble uranium.

It might not be possible to achieve a sensitivity of 5 μg/L (7 E-7 oz/gal); if for example, all reasonable attempts to obtain a 1.4 L (0.37 gal) 24-hour sample within 10 days fail. In such a case, the sample is analyzed for uranium concentration (if measurable) and the worker's intake is estimated using other available data.

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to the internal bioassay program and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994), Section 8.4.1.2.

4.8.1.2 Air Monitoring and Sampling

Airborne activity in work areas is regularly determined in accordance with written procedures. Continuous air sampling in airborne radioactivity areas may be performed to complement the bioassay program. Using the values specified in 10 CFR 20 Appendix B (CFR, 2003m), if a worker could have inhaled radionuclide concentrations that are likely to exceed 12 DAC-hours in one week (seven days), then bioassay is conducted within 72 hours after the suspected or known exposure. Follow-up bioassay measurements are conducted to determine the committed effective dose equivalent. Until urinalysis results indicate less than 15 micrograms per liter uranium concentration, workers are restricted from activities that could routinely or accidentally result in internal exposures to soluble uranium.

Active on-line monitors for airborne alpha emitters are used to measure representative airborne concentrations of radionuclides that may be due to facility operation. On-line monitoring for gross alpha activity is performed assuming all the alpha activity is due to uranium. When airborne activity data is used for dose calculations, the assumption is that all the activity is due to ^{234}U, class D material. The lower limit of detection is either 0.02 mg (7.16 E-7 oz) of uranium in the total sample or 3.7 nBq/mL (1 E-13 μCi/mL) gross alpha concentration. An action level is established at 1 mg (3.53 E-5 oz) of total uranium likely to be inhaled by a worker in seven days.

Monitors are permanently located in Restricted Areas. These permanent monitors are operated to collect continuous samples. When air sampling is conducted using continuous air sampling devices, the filters are changed and analyzed at the following frequencies:

- Weekly and following any indication of release that might lead to airborne concentrations of uranium that are likely to exceed (1) 10% of the values listed in 10 CFR 20.1003 (CFR, 2003n), or (2) the total uranium action level of one milligram of total uranium inhaled in one week.

- Each Shift, following changes in process equipment or process control, and following detection of any event (e.g., leakage, spillage or blockage of process equipment) that are likely to exceed (1) 10% of the values listed in 10 CFR 20.1003 (CFR, 2003n), Airborne Radioactivity Area, or (2) the total uranium action level of one milligram inhaled by a worker in one week.

The representativeness of the workstation air samplers shall be checked annually and when significant process or equipment changes have been made. Facility procedures specify how representativeness is determined.

Plant areas surveyed as described in this section include as a minimum UF_6 processing areas, decontamination areas, waste processing areas and laboratories. Continuous air monitors (e.g., stationary samplers or personnel lapel samplers) may be substituted when appropriate, as when continuous monitoring may not be reasonably achieved.

Action levels are based on trending of data collected during facility operation. Investigations are performed if airborne activity:

A. Exceeds 10% of the values listed in 10 CFR 20.1003 (CFR, 2003n) for Airborne Radioactivity Areas

B. Shows a short-term increase of a factor of 10 over historical data from the previous 12 months.

Corrective actions include investigation of the adverse trend and an evaluation of the need for changes, consistent with the principles of ALARA.

4.8.2 External Exposures

As noted previously, the potential for significant external exposure to personnel under routine operating conditions is less significant than that for internal exposures. This is primarily due to the nature of the radionuclides present in the facility.

Parameters important in determining dose from external exposures are:

- The length of time the worker remains in the radiation field
- The intensity of the radiation field
- The portion of the body receiving the dose.

Historical data from European facilities of similar construction show relatively low doses compared to nuclear power plant doses.

4.8.3 Procedures

Procedures are provided in the following areas to administratively control personnel radiation exposure:

- Operation
- Design
- Maintenance
- Modification
- Decontamination
- Surveillance
- Procurement.

4.8.4 Instrumentation

Two basic types of personnel monitoring equipment are used at the facility. These are count rate meters (as known as "friskers") and hand/foot monitors.

4.8.4.1 Friskers

These typically consist of a hand-held Eberline HP 210/260 (or equivalent) probe connected to a RM-14 (or equivalent) count rate meter. Instructions for the use of these instruments are posted in a prominent location near the instrument. Hand held friskers are typically placed in locations where conditions restrict the use of other monitors or for short-term use as necessary to ensure effective control of the spread of contamination.

4.8.4.2 Hand and Foot Monitors

These typically consist of multiple detectors arranged to monitor only hands and feet. Instructions for the use of these monitors are prominently posted on or near the instrument. Hand and foot monitors are used in applications where "pass-throughs" are frequent and where hand and foot monitoring is the major requirement. Portal monitors, that can quickly scan large surface areas of the body, may be used where the number of personnel exiting an area, available space, etc., makes their use advantageous.

4.8.5 Contamination Control

Small contamination areas (i.e., less than one-fourth of the room) may be roped off or otherwise segregated from the rest of a Restricted Area. Appropriate clothing and/or other equipment is used to minimize exposure to radioactive material and prevent the spread of contamination. Provisions for monitoring contamination and airborne activity levels are discussed below. A contamination monitor (frisker), a step-off pad and a container for any discarded protective clothing may be placed at the access/egress point to the work area. The entire Restricted Area is not posted as a Contaminated Area.

4.8.5.1 Surface Contamination

Contamination survey monitoring is performed for all UF_6 process areas. Surveys include routine checks of non-UF_6 process areas, including areas normally not contaminated. Monitoring includes direct radiation and removable contamination measurements. Survey procedures are based on the potential for contamination of an area and operational experience. The Restricted Areas are surveyed at least weekly. The lunch room and change rooms are surveyed at least daily.

Removable surface contamination is considered uranium contamination that is present on a surface and that can be transferred to a dry smear paper by rubbing with moderate pressure. The facility uses various instruments such as proportional counters, alpha scintillation counters and thin window Geiger-Mueller tubes, to evaluate contamination levels.

Laundered protective clothing is periodically surveyed for gross alpha and gross beta contamination. Levels of less than 2.5 Bq/100 cm^2 (150 dpm/100 cm^2), alpha or beta/gamma are acceptable. This action level should be readily achievable since most of the radioactive material that can contaminate protective clothing at the facility is in soluble form and is easily removed by laundering.

If surface contamination levels exceed the following levels, clean-up of the contamination is initiated within 24 hours of the completion of the analysis:

- Removable contamination: 83.3 Bq/100 cm^2 (5000 dpm/100 cm^2) alpha or beta/gamma
- Fixed contamination: 4.2 kBq/100 cm^2 (250,000 dpm/100 cm^2) alpha or beta/gamma

The subject matter discussed above is identical to Claiborne Enrichment Center SAR (LES, 1993) subject matter. The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) application relative to the surface and personnel contamination control program and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1491 (NRC, 1994), Section 8.4.1.4.

4.9 MAINTENANCE AREAS-METHODS AND PROCEDURES FOR CONTAMINATION CONTROL

Designing processes and equipment that contain radioactive material to require as little maintenance as possible ensures that personnel radiation exposures are ALARA. Additional exposure reductions are achieved by:

A. Removing as much radioactive material as possible from the equipment and the area prior to maintenance, thereby reducing the intensity of the radiation field

B. Providing adequate space for ease of maintenance reducing the length of time required to complete the task, thereby reducing the time of exposure

C. Preparing and using procedures that contain specifications for tools and equipment needed to complete the job

D. Proper job planning, including practice on mockups

E. Previews of previous similar jobs

F. Identification and communication of the highest contamination areas to the workers prior to the start of work.

4.9.1 Decontamination Workshop

The Contaminated Workshop and Decontamination System are located in the same room in the TSB. This room is called the Decontamination Workshop. The Decontamination Workshop in the TSB contains an area to break down and strip contaminated equipment and to decontaminate the equipment and its components. The decontamination systems in the workshop are designed to remove radioactive contamination from contaminated materials and equipment. The only significant forms of radioactive contamination found in the facility are uranium hexafluoride (UF_6), uranium tetrafluoride (UF_4) and uranyl fluoride (UO_2F_2).

One of the functions of the Decontamination Workshop is to provide a maintenance facility for both UF_6 pumps and for vacuum pumps. The workshop is used for the temporary storage and subsequent dismantling of failed pumps. The dismantling area is in physical proximity to the decontamination train, in which the dismantled pump components are processed.

The process carried out within the Decontamination Workshop begins with receipt and storage of contaminated pumps, out-gassing, Fomblin oil removal and storage, and pump stripping. Activities for the dismantling and maintenance of other plant components are also carried out. Other components commonly decontaminated besides pumps include valves, piping, instruments, sample bottles, tools, and scrap metal. Personnel entry into the facility is via a sub-change facility. This area has the required contamination area access controls, washing and monitoring facilities.

The decontamination part of the process consists of a series of steps following equipment disassembly including degreasing, decontamination, drying, and inspection. Items from uranium hexafluoride systems, waste handling systems, and miscellaneous other items are decontaminated in this system.

4.9.2 Laundry System

The Laundry System cleans contaminated and soiled clothing and other articles which have been used throughout the plant. It contains the resulting solid and liquid wastes for transfer to appropriate treatment and disposal facilities. The Laundry System receives the clothing and articles from the plant in plastic bin bags, taken from containers strategically positioned within the plant. Clean clothing and articles are delivered to storage areas located within the plant. The Laundry System components are located in the Laundry room of the TSB.

The Laundry System collects, sorts, cleans, dries, and inspects clothing and articles used in Restricted Areas of the plant. Laundry collection is divided into two main groups; articles with a low probability of contamination and articles with a high probability of contamination. Those articles unlikely to have been contaminated are further sorted into lightly soiled and heavily soiled groups. The sorting is done on a table underneath a vent hood that is connected to the TSB GEVS. All lightly soiled articles are cleaned in the laundry. Heavily soiled articles are inspected and any considered to be difficult to clean (i.e., those with significant amounts of grease or oil on them) are transferred to the Solid Waste Collection System without cleaning. Articles from one plant department are not cleaned with articles from another plant department.

Special water-absorbent bags are used to collect the articles that are more likely to be contaminated. These articles may include pressure suits and items worn when, for example, it is required to disconnect or "open up" an existing plant system. These articles that are more likely to be contaminated are cleaned separately. Expected contaminants on the laundry include slight amounts of uranyl fluoride (UO_2F_2) and uranium tetrafluoride (UF_4).

When sorting is completed, the articles are placed in a washing machine in batches. No "dry cleaning" solvents are used. Wastewater from the washing machine is discharged to one of three Laundry Effluent Monitor Tanks in the Liquid Effluent Collection and Treatment System. The laundry effluent is then sampled, analyzed, and transferred to the Treated Effluent Evaporative Basin or to the Precipitation Treatment Tank for additional treatment as necessary.

When the washing cycle is complete, the wet laundry is placed in an electrically heated dryer. The dryer has variable temperature settings, and the hot wet air is exhausted to the atmosphere through a lint drawer that is built into the dryer. The lint from the drawer is then sent to the Solid Waste Collection System as combustible waste. Dry laundry is removed from the dryer and placed on the laundry inspection table for inspection and folding. Folded laundry is returned to storage areas in the plant.

4.10 DECONTAMINATION POLICY AND PROVISIONS

Removing radioactive material from equipment, to the extent reasonably possible prior to servicing, reduces exposures to personnel who work around and service contaminated equipment. Surface contamination is removed to minimize its spread to other areas of the facility. Surfaces such as floors and walls are designed to be smooth, nonporous and free of cracks so that they can be more easily decontaminated.

Decontamination facilities and procedures for the Technical Services Building and the Separations Building Module have been discussed above. For the remaining areas of the Separations Building Module, decontamination requirements involve only localized clean-up at areas where maintenance has been or is being performed that involves opening a uranium-containing system. All decontamination of components removed from their systems for maintenance is performed in Technical Services Building. No other areas of the facility normally require decontamination.

The facility follows NRC Branch Technical Position: Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material (NRC, 1993e). This guide applies to the abandonment or release for unrestricted use, of surfaces, premises and equipment.

4.11 ADDITIONAL PROGRAM COMMITMENTS

The following section describes additional program commitments related to the Radiation Protection Program.

4.11.1 Leak-Testing Byproduct Material Sources

In addition to the uranium processed at the facility, other sources of radioactivity are used. These sources are small calibration sources used for instrument calibration and response checking. These byproduct material sources may be in solid, liquid, or gaseous form; the sources may be sealed or unsealed. Both types of sources present a small radiation exposure risk to facility workers. Typical byproduct material quantities and uses for a Urenco uranium enrichment centrifuge plant are summarized in Table 4.11-1, Typical Quantities of Byproduct Material for a Urenco Uranium Enrichment Centrifuge Plant. The byproduct materials for the NEF will be identified during the design phase and the Safety Analysis Report will be revised accordingly. Leak-testing of sources is performed in accordance with the following NRC Branch Technical Positions (BTPs):

A. License Condition for Leak-Testing Sealed Byproduct Material Sources (NRC,1993b)

B. License Condition for Leak-Testing Sealed Source Which Contains Alpha and/or Beta-Gamma Emitters (NRC, 1993c)

C. License Condition for Leak-Testing Sealed Uranium Sources (NRC, 1993d)

The following BTPs were not included in this section since the facility has not requested sources containing plutonium (refer to Table 4.11-1):

- *License Condition for Leak-Testing Sealed Plutonium Sources*, April 1993
- *License Condition for Plutonium Alpha Sources,* April 1993.

4.11.2 Records and Reports

The facility meets the following regulations for the additional program commitments applicable to records and reports:

- 10 CFR 20 (CFR, 2003b), Subpart L (Records), Subpart M (Reports)
- Section 70.61 (Performance requirements) (CFR, 2003e)
- Section 70.74 (Additional reporting requirements) (CFR, 2003s).

The facility Records Management program is described in Section 11.7, Records Management. The facility maintains complete records of the Radiation Protection Program for at least the life of the facility.

The facility maintains records of the radiation protection program (including program provisions, audits, and reviews of the program content and implementation), radiation survey results (air sampling, bioassays, external-exposure data from monitoring of individuals, internal intakes of radioactive material), and results of corrective action program referrals, RWPs and planned special exposures.

By procedure, the facility will report to the NRC, within the time specified in 10 CFR 20.2202 (CFR, 2003t) and 10 CFR 70.74 (CFR, 2003s), any event that results in an occupational exposure to radiation exceeding the dose limits in 10 CFR 20 (CFR, 2003b). The facility will prepare and submit to the NRC an annual report of the results of individual monitoring, as required by 10 CFR 20.2206(b) (CFR, 2003u).

As previously noted in this chapter, LES will refer to the facility's corrective action program any radiation incident that results in an occupational exposure that exceeds the dose limits in 10 CFR 20, Appendix B (CFR, 2003m), or is required to be reported per 10 CFR 70.74 (CFR, 2003s). The facility reports to the NRC both the corrective action taken (or planned) to protect against a recurrence and the proposed schedule to achieve compliance with the applicable license condition or conditions.

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TABLES

Table 4.1-1 Administrative Radiation Exposure Limits
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	Administrative Limit
Total Effective Dose Equivalent (TEDE)	10 mSv/yr (1000 mrem/yr)

Notes:

a) Excludes accident situations

b) No routine extremity or skin monitoring is required

c) TEDE is the sum of internal dose and external dose received during routine operations

d) NRC limit is 50 mSv/yr (5000 mrem/yr)

Table 4.1-2 Estimated Dose Rates

Area or Component	Dose Rate, mSv/hr (mrem/hr)
Plant general area (excluding Separations Building Module)	< 1 E-4 (< 0.01)
Separations Building Module – Cascade Halls	5 E-4 (0.05)
Separations Building Module –UF_6 Handling Area & Process Services Area	1 E-3 (0.1)
Empty used UF_6 shipping cylinder	0.1 on contact (10.0) 0.01 at 1 m (1.0)
Full UF_6 shipping cylinder	0.05 on contact (5.0) 2 E-3 at 1 m (0.2)

Table 4.1-3 Estimated Individual Exposures

Page 1 of 1

Position	Annual Dose [(a)] mSv (mrem)
General Office Staff	< 0.05 (< 5.0)
Typical Operations & Maintenance Technician	1 (100)
Typical Cylinder Handler	3 (300)

(a) The average worker exposure at the Urenco Capenhurst facility during the years 1998 through 2002 was approximately 0.2 mSv (20 mrem) (Urenco, 2000; Urenco, 2001; Urenco, 2002)

Table 4.7-1 Radiation Emitted from Natural UF_6 Feed

Element	Nuclide Symbol	Half-Life	Maximum Radiation Energies (Mev) and Intensities		
			alpha (α)	beta (β)	gamma (γ)
92 uranium	^{238}U	4.5E+9 yr	4.15 25% 4.20 75%	none	0.013 8.8%
90 thorium	^{231}Th	26 hr	none	0.39 ~100%	0.025 14.7%
90 thorium	^{234}Th	24 d	none	0.19 73% 0.10 27%	0.06 3.8% 0.09 5.4%
91 protactinium	^{234}Pa	1.2 min	none	2.28 99%	0.766 0.21% 1.001 0.60%
92 uranium	^{234}U	2.5E+5 yr	4.72 28% 4.78 72%	none	0.053 0.12%
92 uranium	^{235}U	7.04E+8 yr	4.37 17% 4.40 55% 4.60 14%	none	0.143 12% 0.185 54% 0.205 6%

Table 4.11-1 Typical Quantities of Byproduct Material for a Urenco Uranium Enrichment Centrifuge Plant

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Radionuclide	Quantity	Use
^{3}H	19 GBq (5.14E-01 Ci)	Instrument calibration or response checking
^{36}Cl	8.35 kBq (2.26E-07 Ci)	Instrument calibration or response checking
^{57}Co	930 MBq (2.51E-02 Ci)	Instrument calibration or response checking
^{90}Sr	1.04kBq (2.81E-08 Ci)	Instrument calibration or response checking
^{99}Tc	3.09 kBq (8.35E-08 Ci)	Instrument calibration or response checking
^{109}Cd	37 MBq (1.00E-03 Ci)	Instrument calibration or response checking
^{131}Cs	390 Bq (1.05E-08 Ci)	Instrument calibration or response checking
^{133}Ba	0.7 MBq (1.89E-05 Ci)	Instrument calibration or response checking
^{137}Cs	2.05 GBq (5.53E-02 Ci)	Instrument calibration or response checking
^{210}Po	63 MBq (1.70E-03 Ci)	Instrument calibration or response checking
^{226}Ra	38 MBq (1.03E-03 Ci)	Instrument calibration or response checking
^{233}U	3.7 GBq (1.00E-01 Ci)	Instrument calibration or response checking
^{234}U	4.4 Bq (1.19E-10 Ci)	Instrument calibration or response checking
^{235}U	3.7 GBq (1.00E-01 Ci)	Instrument calibration or response checking
^{236}U	3.7 GBq (1.00E-01 Ci)	Instrument calibration or response checking
^{237}Np	2.0 kBq (5.41E-08 Ci)	Instrument calibration or response checking
^{238}U	164.5 Bq (4.45E-09 Ci)	Instrument calibration or response checking
^{241}Am	1.1GBq (2.97E-02 Ci)	Instrument calibration or response checking

Byproduct material may be in solid, liquid, or gaseous form. Byproduct material is not necessarily restricted to sealed sources.

FIGURES

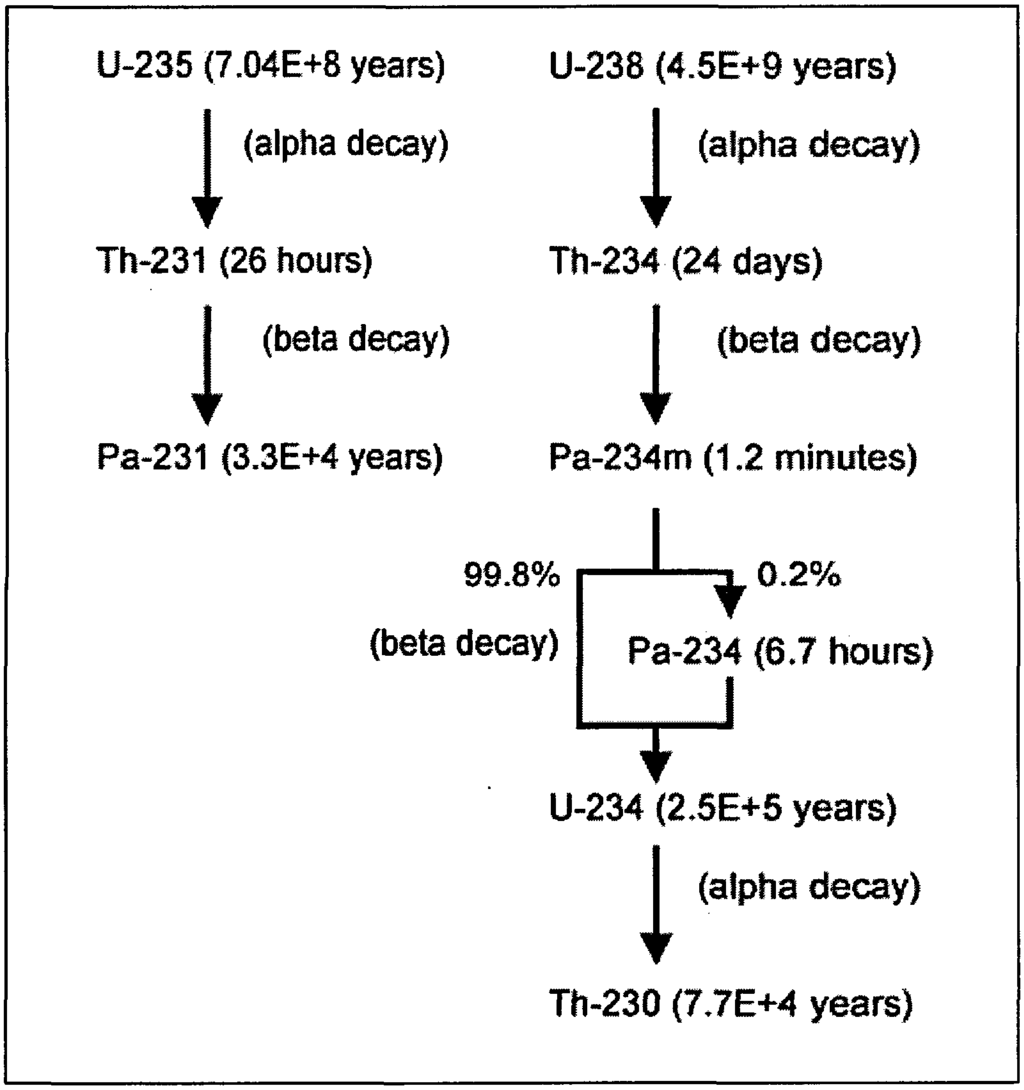
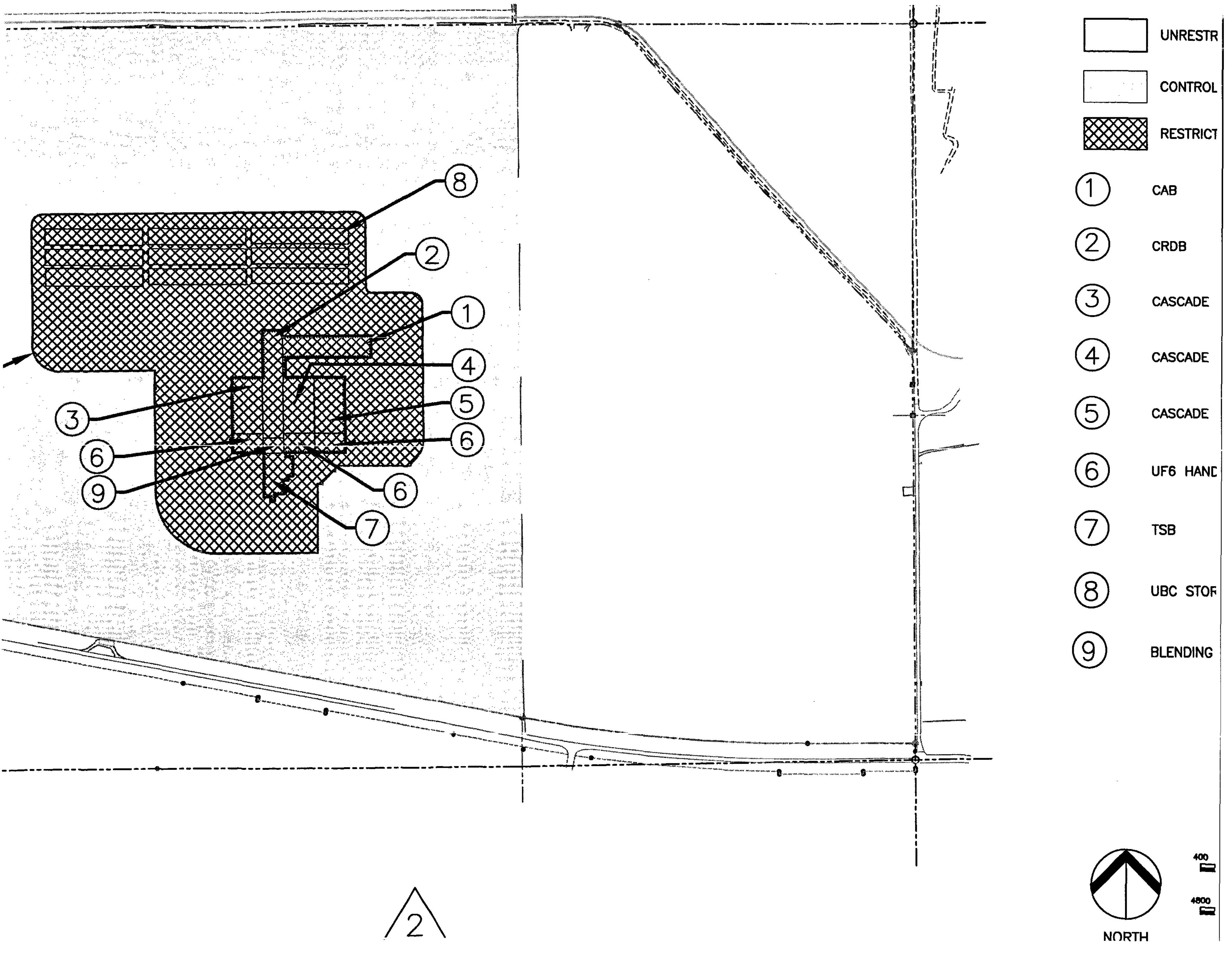


FIGURE 4.7-1
URANIUM AND DECAY PRODUCTS OF INTEREST

REFERENCE NUMBER Figure 4.7-1.doc

REVISION DATE: DECEMBER 2003



UNRESTR
CONTROL
RESTRICT
1 CAB
2 CRDB
3 CASCADE
4 CASCADE
5 CASCADE
6 UF6 HAND
7 TSB
8 UBC STOF
9 BLENDING
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5.0 NUCLEAR CRITICALITY SAFETY

The Nuclear Criticality Safety Program for the National Enrichment Facility (NEF) is in accordance with U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 3.71, Nuclear Criticality Safety Standards for Fuels and Material Facilities (NRC, 1998). Regulatory Guide 3.71 (NRC, 1998) provides guidance on complying with the applicable portions of NRC regulations, including 10 CFR 70 (CFR, 2003a), by describing procedures for preventing nuclear criticality accidents in operations involving handling, processing, storing, and transporting special nuclear material (SNM) at fuel and material facilities. The facility is committed to following the guidelines in this regulatory guide for specific ANSI/ANS criticality safety standards with the exception of ANSI/ANS-8.9-1987, "Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Material." Piping configurations containing aqueous solutions of fissile material will be evaluated in accordance with ANSI/ANS-8.1-1998 (ANSI, 1998a), using validated methods to determine subcritical limits.

The information provided in this chapter, the corresponding regulatory requirements, and the section of NUREG-1520 (NRC, 2002), Chapter 5 in which the NRC acceptance criteria are presented is summarized below.

Information Category and Requirement	10 CFR 70 Citation	NUREG-1520 Chapter 5 Reference
Section 5.1 Nuclear Criticality Safety (NCS) Program		
Management of the NCS Program	70.61(d) 70.64(a)	5.4.3.1
Control Methods for Prevention of Criticality	70.61	5.4.3.4.2
Safe Margins Against Criticality	70.61	5.4.3.4.2
Description of Safety Criteria	70.61	5.4.3.4.2
Organization and Administration	70.61	5.4.3.2
Section 5.2 Methodologies and Technical Practices		
Methodology	70.61	5.4.3.4.1 5.4.3.4.4 5.4.3.4.6
Section 5.3 Criticality Accident Alarm System (CAAS)		
Criticality Accident Alarm System	70.24	5.4.3.4.3
Section 5.4 Reporting		
Reporting Requirements	Appendix A	5.4.3.4.7 (7)

5.1 THE NUCLEAR CRITICALITY SAFETY (NCS) PROGRAM

The facility has been designed and will be constructed and operated such that a nuclear criticality event is prevented, and to meet the regulatory requirements of 10 CFR 70 (CFR, 2003a). Nuclear criticality safety at the facility is assured by designing the facility, systems and components with safety margins such that safe conditions are maintained under normal and abnormal process conditions and any credible accident. Items Relied On For Safety (IROFS) identified to ensure subcriticality are discussed in the NEF Integrated Safety Analysis Summary.

5.1.1 Management of the Nuclear Criticality Safety (NCS) Program

The NCS criteria in Section 5.2, Methodologies and Technical Practices, are used for managing criticality safety and include adherence to the double contingency principle as stated in the ANSI/ANS-8.1-1998, Nuclear Criticality Safety In Operations with Fissionable Materials Outside Reactors (ANSI, 1998a). The adopted double contingency principle states "process design should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible." Each process that has accident sequences that could result in an inadvertent nuclear criticality at the NEF meets the double contingency principle. The NEF meets the double contingency principle in that process design incorporates sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

Using these NCS criteria, including the double contingency principle, low enriched uranium enrichment facilities have never had an accidental criticality. The plant will produce no greater than 5.0 $^{w}/_{o}$ enrichment. However, as additional conservatism, the nuclear criticality safety analyses are performed assuming a ^{235}U enrichment of 6.0 $^{w}/_{o}$, except for Contingency Dump System traps which are analyzed assuming a ^{235}U enrichment of 1.5 $^{w}/_{o}$, and include appropriate margins to safety. In accordance with 10 CFR 70.61(d) (CFR, 2003b), the general criticality safety philosophy is to prevent accidental uranium enrichment excesses, provide geometrical safety when practical, provide for moderation controls within the UF_6 processes and impose strict mass limits on containers of aqueous, solvent based, or acid solutions containing uranium. Interaction controls provide for safe movement and storage of components. Plant and equipment features assure prevention of excessive enrichment. The plant is divided into six distinctly separate Assay Units (called Cascade Halls) with no common UF_6 piping. UF_6 blending is done in a physically separate portion of the plant. Process piping, individual centrifuges and chemical traps other than the contingency dump chemical traps, are safe by limits placed on their diameters. Product cylinders rely upon uranium enrichment, moderation control and mass limits to protect against the possibility of a criticality event. Each of the liquid effluent collection tanks that hold uranium in solution is mass controlled, as none are geometrically safe. As required by 10 CFR 70.64(a) (CFR, 2003c), by observing the double contingency principle throughout the plant, a criticality accident is prevented. In addition to the double contingency principle, effective management of the NCS Program includes:

- An NCS program to meet the regulatory requirements of 10 CFR 70 (CFR, 2003a) will be developed, implemented, and maintained.

- Safety parameters and procedures will be established.
- The NCS program structure, including definition of the responsibilities and authorities of key program personnel will be provided.
- The NCS methodologies and technical practices will be kept applicable to current configuration by means of the configuration management function. The NCS program will be upgraded, as necessary, to reflect changes in the ISA or NCS methodologies and to modify operating and maintenance procedures in ways that could reduce the likelihood of occurrence of an inadvertent nuclear criticality.
- The NCS program will be used to establish and maintain NCS safety limits and NCS operating limits for IROFS in nuclear processes and a commitment to maintain adequate management measures to ensure the availability and reliability of the IROFS.
- NCS postings will be provided and maintained current.
- NCS emergency procedure training will be provided.
- The NCS baseline design criteria requirements in 10 CFR 70.64(a) (CFR, 2003c) will be adhered to.
- The NCS program will be used to evaluate modifications to operations, to recommend process parameter changes necessary to maintain the safe operation of the facility, and to select appropriate IROFS and management measures.
- The NCS program will be used to promptly detect NCS deficiencies by means of operational inspections, audits, and investigations. Deficiencies will be entered into the corrective action program so as to prevent recurrence of unacceptable performance deficiencies in IROFS, NCS function or management measures.
- NCS program records will be retained as described in Section 11.7, Records Management.

Training will be provided to individuals who handle nuclear material at the facility in criticality safety. The training is based upon the training program described in ANSI/ANS-8.20-1991, Nuclear Criticality Safety Training (ANSI, 1991). The training program is developed and implemented with input from the criticality safety staff, training staff, and management. The training focuses on the following:

- Appreciation of the physics of nuclear criticality safety.
- Analysis of jobs and tasks to determine what a worker must know to perform tasks efficiently.
- Design and development of learning objectives based upon the analysis of jobs and tasks that reflect the knowledge, skills, and abilities needed by the worker.
- Implementation of revised or temporary operating procedures.

Additional discussion of management measures is provided in Chapter 11, Management Measures.

5.1.2 Control Methods for Prevention of Criticality

The major controlling parameters used in the facility are enrichment control, geometry control, moderation control, and/or limitations on the mass as a function of enrichment. In addition, reflection, interaction, and heterogeneous effects are important parameters considered and applied where appropriate in nuclear criticality safety analyses. Nuclear Criticality Safety Evaluations and Analyses are used to identify the significant parameters affected within a particular system. All assumptions relating to process, equipment, material function, and operation, including credible abnormal conditions, are justified, documented, and independently reviewed. Where possible, passive engineered controls are used to ensure NCS. The determination of the safe values of the major controlling parameters used to control criticality in the facility is described below.

Moderation control is in accordance with ANSI/ANS-8.22-1997, Nuclear Criticality Safety Based on Limiting and Controlling Moderators (ANSI, 1997). However, for the purposes of the criticality analyses, it is assumed that UF_6 comes in contact with water to produce aqueous solutions of UO_2F_2 as described in Section 5.2.1.3.3, Uranium Accumulation and Moderation Assumption. A uniform aqueous solution of UO_2F_2, and a fixed enrichment are conservatively modeled using MONK8A (SA, 2001) and the JEF2.2 library. Criticality analyses were performed to determine the maximum value of a parameter to yield $k_{eff} = 1$. The criticality analyses were then repeated to determine the maximum value of the parameter to yield a $k_{eff} = 0.95$. Table 5.1-1, Safe Values for Uniform Aqueous Solution of Enriched UO_2F_2, shows both the critical and safe limits for 5.0 $^w/_o$ and 6.0 $^w/_o$.

Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, lists the safety criteria of Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2, which are used as control parameters to prevent a nuclear criticality event. Although the NEF will be limited to 5.0 $^w/_o$ enrichment, as additional conservatism, the values in Table 5.1-2, Safety Criteria for Buildings/Systems/ Components, represent the limits based on 6.0 $^w/_o$ enrichment except for the Contingency Dump System traps which are limited to 1.5 $^w/_o$ ^{235}U.

The values on Table 5.1-1 are chosen to be critically safe when optimum light water moderation exists and reflection is considered within isolated systems. The conservative modeling techniques provide for more conservative values than provided in ANSI/ANS-8.1 (ANSI, 1998a). The product cylinders are only safe under conditions of limited moderation and enrichment. In such cases, both design and operating procedures are used to assure that these limits are not exceeded.

All Separation Plant components, which handle enriched UF_6, other than the Type 30B and 48Y cylinders and the first stage UF_6 pumps and contingency dump chemical traps, are safe by geometry. Centrifuge array criticality is precluded by a probability argument with multiple operational procedure barriers. Total moderator or H/U ratio control as appropriate precludes product cylinder criticality.

In the Technical Services Building (TSB) criticality safety for uranium loaded liquids is ensured by limiting the mass of uranium in any single tank to less than or equal to 12.2 kg U (26.9 lb U). Individual liquid storage bottles are safe by volume. Interaction in storage arrays is accounted for.

Based on the criticality analyses, the control parameters applied to NEF are as follows:

Enrichment

Enrichment is controlled to limit the percent ^{235}U within any process, vessel, or container, except the contingency dump system, to a maximum enrichment of 5 $^{w}/_{o}$. The design of the contingency dump system controls enrichment to a limit of 1.5 $^{w}/_{o}$ ^{235}U. Although NEF is limited to a maximum enrichment of 5 $^{w}/_{o}$, as added conservatism nuclear criticality safety is analyzed using an enrichment of 6 $^{w}/_{o}$ ^{235}U.

Geometry/Volume

Geometry/volume control may be used to ensure criticality safety within specific process operations or vessels, and within storage containers.

The geometry/volume limits are chosen to ensure k_{eff} ($k_{calc} + 3\ \sigma_{calc}$) $\leq$ 0.95.

The safe values of geometry/volume define the characteristic dimension of importance for a single unit such that nuclear criticality safety is not dependent on any other parameter assuming 6 $^{w}/_{o}$ ^{235}U for safety margin.

Moderation

Water and oil are the moderators considered in NEF. At NEF the only system where moderation is used as a control parameter is in the product cylinders. Moderation control is established consistent with the guidelines of ANSI/ANS-8.22-1997 (ANSI, 1997) and incorporates the criteria below:

- Controls are established to limit the amount of moderation entering the cylinders.
- When moderation is the only parameter used for criticality control, the following additional criteria are applied. These controls assure that at least two independent controls would have to fail before a criticality accident is possible.
  - Two independent controls are utilized to verify cylinder moderator content.
  - These controls are established to monitor and limit uncontrolled moderator prior to returning a cylinder to production thereby limiting the amount of uncontrolled moderator from entering a system to an acceptable limit.
  - The evaluation of the cylinders under moderation control includes the establishment of limits for the ratio of maximum moderator-to-fissile material for both normal operating and credible abnormal conditions. This analysis has been supported by parametric studies.
- When moderation is not considered a control parameter, either optimum moderation or worst case H/U ratio is assumed when performing criticality safety analysis.

Mass

other control methods. Analysis or sampling is employed to verify the mass of the material. Conservative administrative limits for each operation are specified in the operating procedures.

Whenever mass control is established for a container, records are maintained for mass transfers into and out of the container. Establishment of mass limits for a container involves consideration of potential moderation, reflection, geometry, spacing, and enrichment. The evaluation considers normal operations and credible abnormal conditions for determination of the operating mass limit for the container and for the definition of subsequent controls necessary to prevent reaching the safety limits. When only administrative controls are used for mass controlled systems, double batching is conservatively assumed in the analysis.

Reflection

Reflection is considered when performing Nuclear Criticality Safety Evaluations and Analyses. The possibility of full water reflection is considered but the layout of the NEF is a very open design and it is highly unlikely that those vessels and plant components requiring criticality control could become flooded from a source of water within the plant. In addition, neither automatic sprinkler nor standpipe and hose systems are provided in the TSB, Separation Buildings, Blending and Liquid Sampling, CRDB, CAB, and Centrifuge Post Mortem areas. Therefore, full water reflection of vessels has therefore been discounted. However, some select analyses have been performed using full reflection for conservatism. Partial reflection of 2.5 cm (0.984 in) of water is assumed where limited moderating materials (including humans) may be present It is recognized that concrete can be a more efficient reflector than water; therefore, it is modeled in analyses where it is present. When moderation control is identified in the ISA Summary, it is established consistent with the guidelines of ANSI/ANS-8.22-1997 (ANSI, 1997).

Interaction

Nuclear criticality safety evaluations and analyses consider the potential effects of interaction. A non-interacting unit is defined as a unit that is spaced an approved distance from other units such that the multiplication of the subject unit is not increased. Units may be considered non-interacting when they are separated by more than 60 cm (23.6 inches).

If a unit is considered interacting, nuclear criticality safety analyses are performed. Individual unit multiplication and array interaction are evaluated using the Monte Carlo computer code MONK8A to ensure k_{eff} (k_{calc} + 3 σ_{calc}) < 0.95.

Concentration, Density and Neutron Absorbers

NEF does not use mass concentration, density, or neutron absorbers as a criticality control parameter.

5.1.3 Safe Margins Against Criticality

Process operations require establishment of criticality safety limits. The facility UF_6 systems involve mostly gaseous operations. These operations are carried out under reduced atmospheric conditions (vacuum) or at slightly elevated pressures not exceeding three atmospheres. It is highly unlikely that any size changes of process piping, cylinders, cold traps, or chemical traps under these conditions, would lead to a criticality situation because a volume or mass limit may be exceeded.

Within the Separations Building, significant accumulations of enriched UF_6 reside only in the Product Low Temperature Take-off Stations, Product Liquid Sampling Autoclaves, Product Blending System or the UF_6 cold traps. All these, except the UF_6 cold traps, contain the UF_6 in 30B and 48Y cylinders. All these significant accumulations are within enclosures protecting them from water ingress. The facility design has minimized the possibility of accidental moderation by eliminating direct water contact with these cylinders of accumulated UF_6. In addition, the facility's stringent procedural controls for enriching the UF_6 assure that it does not become unacceptably hydrogen moderated while in process. The plant's UF_6 systems operating procedures contain safeguards against loss of moderation control (ANSI, 1997). No neutron poisons are relied upon to assure criticality safety.

5.1.4 Description of Safety Criteria

Each portion of the plant, system, or component that may possibly contain enriched uranium is designed with criticality safety as an objective. Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, shows how the safety criteria of Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2, are applied to the facility to prevent a nuclear criticality event. Although the NEF will be limited to 5.0 $^{w}/_{o}$ enrichment, as additional conservatism, the values in Table 5.1-2, represent the limits based on 6.0 $^{w}/_{o}$ enrichment.

Where there are significant in-process accumulations of enriched uranium as UF_6, the plant design includes multiple features to minimize the possibilities for breakdown of the moderation control limits. These features eliminate direct ingress of water to product cylinders while in process.

5.1.5 Organization and Administration

The criticality safety organization is responsible for implementing the Nuclear Criticality Safety Program. During the design phase, the criticality safety function is performed within the design engineering organization. The criticality safety function for operations is described in the following section.

The criticality safety organization reports to the Health, Safety, and Environment (HS&E) Manager as described in Chapter 2, Organization and Administration. The HS&E Manager is accountable for overall criticality safety of the facility, is administratively independent of production responsibilities, and has the authority to shut down potentially unsafe operations.

Designated responsibilities of the criticality safety staff include the following:

- Establish the Nuclear Criticality Safety Program, including design criteria, procedures, and training
- Provide criticality safety support for integrated safety analyses and configuration control
- Assess normal and credible abnormal conditions
- Determine criticality safety limits for controlled parameters
- Develop and validate methods to support nuclear criticality safety evaluations (NCSEs) (i.e., non-calculation engineering judgments regarding whether existing criticality safety analyses bound the issue being evaluated or whether new or revised safety analyses are required)

- Perform NCS analyses (i.e., calculations), write NCS evaluations, and approve proposed changes in process conditions on equipment involving fissionable material
- Specify criticality safety control requirements and functionality
- Provide advice and counsel on criticality safety control measures, including review and approval of operating procedures
- Support emergency response planning and events
- Evaluate the effectiveness of the Nuclear Criticality Safety Program using audits and assessments
- Provide criticality safety postings that identify administrative controls for operators in applicable work areas.

The minimum qualifications for a criticality safety engineer are a Bachelor of Science (BS) or Bachelor of Arts (BA) degree in science or engineering with at least two years of nuclear industry experience in criticality safety. A criticality safety engineer must understand and have experience in the application and direction of criticality safety programs. The HS&E Manager has the authority and responsibility to assign and direct activities for the criticality safety staff. The criticality safety engineer is responsible for implementation of the NCS program. Criticality safety engineers will be provided in sufficient numbers to implement and support the operation of the NCS program.

The NEF implements the intent of the administrative practices for criticality safety, as contained in Section 4.1.1 of American National Standards Institute/American Nuclear Society (ANSI/ANS)-8.1-1998, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (ANSI, 1998a). A policy will be established whereby personnel shall report defective NCS conditions and perform actions only in accordance with written, approved procedures. Unless a specific procedure deals with the situation, personnel shall report defective NCS conditions and take no action until the situation has been evaluated and recovery procedures provided.

5.2 METHODOLOGIES AND TECHNICAL PRACTICES

This section describes the methodologies and technical practices used to perform the Nuclear Criticality Safety (NCS) analyses and NCS evaluations. The determination of the NCS controlled parameters and their application and the determination of the NCS limits on IROFS are also presented.

5.2.1 Methodology

MONK8A (SA, 2001) is a powerful Monte Carlo tool for nuclear criticality safety analysis. The advanced geometry modeling capability and detailed continuous energy collision modeling treatments provide realistic 3-dimensional models for an accurate simulation of neutronic behavior to provide the best estimate neutron multiplication factor, k-effective. Complex models can be simply set up and verified. Additionally, MONK8A (SA, 2001) has demonstrable accuracy over a wide range of applications and is distributed with a validation database comprising critical experiments covering uranium, plutonium and mixed systems over a wide range of moderation and reflection. The experiments selected are regarded as being representative of systems that are widely encountered in the nuclear industry, particularly with respect to chemical plant operations, transportation and storage. The validation database is subject to on-going review and enhancement. A categorization option is available in MONK8A (SA, 2001) to assist the criticality analyst in determining the type of system being assessed and provides a quick check that a calculation is adequately covered by validation cases.

5.2.1.1 Methods Validation

The validation process establishes method bias by comparing measured results from laboratory critical experiments to method-calculated results for the same systems. The verification and validation processes are controlled and documented. The validation establishes a method bias by correlating the results of critical experiments with results calculated for the same systems by the method being validated. Critical experiments are selected to be representative of the systems to be evaluated in specific design applications. The range of experimental conditions encompassed by a selected set of benchmark experiments establishes the area of applicability over which the calculated method bias is applicable. Benchmark experiments are selected that resemble as closely as practical the systems being evaluated in the design application.

The extensive validation database contains a number of solution experiments applicable to this application involving both low and high-enriched uranium. The MONK8A (SA, 2001) code with the JEF2.2 library was validated against these experiments which are provided in the International Handbook of Evaluated Criticality Safety Benchmark Experiments (NEA, 2002) and Nuclear Science and Engineering (NSE, 1962). The experiments chosen are provided in Table 5.2-1, Uranium Solution Experiments Used for Validation, along with a brief description. The overall mean calculated value from the 80 configurations is 1.0017 ± 0.0005 (AREVA, 2004) and the results are shown in Figure 5.2-1,Validation Results for Uranium Solutions, plotted against H/U-fissile ratio. If only the 36 low-enriched solutions are considered, the mean calculated value is 1.0007 ± 0.0005.

MONK8A is distributed in ready-to-run executable form. This approach provides the user with a level of quality assurance consistent with the needs of safety analysis. The traceability from source code to executable code is maintained by the code vendor. The MONK8A software package contains a set of validation analyses which can be used to support the specific applications. Since the source code is not available to the user, the executable code is identical to that used for the validation analyses. The criticality analyses were performed with MONK8A utilizing the validation provided by the code vendor.

In accordance with the guidance in NUREG-1520 (NRC, 2002), code validation for the specific application has been performed (AREVA, 2004). Specifically, the experiments provided in Table 5.2-1, Uranium Solution Experiments Used for Validation, were calculated and documented as part of the integrated safety analysis for the National Enrichment Facility. In addition, the details of validation should state computer codes used, operations, recipes for choosing code options (where applicable), cross sections sets, and any numerical parameters necessary to describe the input. Therefore, by December 30, 2005, Louisiana Energy Services (LES) will provide NRC with a revised validation report that meets the LES commitment to ANSI/ANS-8.1-1998 (ANSI. 1998a) and includes details of validation that state computer codes used, operations, recipes for choosing code options (where applicable), cross sections sets, and any numerical parameters necessary to describe the input.

The MONK8A computer code and JEF2.2 library are within the scope of the Quality Assurance Program.

5.2.1.2 Limits on Control and Controlled Parameters

The validation process established a bias by comparing calculations to measured critical experiments. With the bias determined, an upper safety limit (USL) can be determined using the following equation from NUREG/CR-6698, Guide for Validation of Nuclear Criticality Safety Calculational Methodology (NRC, 2001):

$$USL = 1.0 + Bias - \sigma_{Bias} - \Delta_{SM} - \Delta_{AOA}$$

Where the critical experiments are assumed to have a k_{eff} of unity, and the bias was determined by comparison of calculation to experiment. From Section 5.2.1.1, Methods Validation, the bias is positive and since a positive bias may be non-conservative, the bias is set to zero. The σ_{Bias} from Section 5.2.1.1, Methods Validation is 0.0005 and a value of 0.05 is assigned to the subcritical margin, Δ_{SM}. The term Δ_{AOA} is an additional subcritical margin to account for extensions in the area of applicability. Since the experiments in the benchmark are representative of the application, the term Δ_{AOA} is set to zero. Thus, the USL becomes:

$$USL = 1 - 0.0005 - 0.05 = 0.9495$$

NUREG/CR-6698 (NRC, 2001) requires that the following condition be demonstrated for all normal and credible abnormal operating conditions:

$$k_{calc} + 2\,\sigma_{calc} < USL$$

In the NCS analysis, σ_{calc} is shown to be greater than σ_{Bias}; therefore, the NEF will be designed using the more conservative equation:

$$k_{eff} = k_{calc} + 3\,\sigma_{calc} < 0.95$$

Additionally, criticality safety in the NEF is ensured by use of geometry, volume, mass and moderation control. Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2 provides the safe values of geometry, volume and mass at 5.0 $^w/_o$ enrichment UO_2F_2 to ensure the USL is met. Moreover, Table 5.1-2, Safety Criteria for Buildings/Systems/Components, provides the additional conservatism used in the design of the NEF. All criticality safety analyses use an enrichment of 6.0 $^w/_o$ ^{235}U, except for Contingency Dump System traps which are analyzed using an enrichment of 1.5 $^w/_o$ ^{235}U, while the facility is limited to an enrichment of 5.0 $^w/_o$ ^{235}U.

5.2.1.3 General Nuclear Criticality Safety Methodology

The NCS analyses results provide values of k-effective (k_{eff}) to conservatively meet the upper safety limit. The following sections provide a description of the major assumptions used in the NCS analyses.

5.2.1.3.1 Reflection Assumption

The layout of the NEF is a very open design and it is not considered credible that those vessels and plant components requiring criticality control could become flooded from a source of water within the plant. Full water reflection of vessels has therefore been discounted. However, where appropriate, spurious reflection due to walls, fixtures, personnel, etc. has been accounted for by assuming 2.5 cm (0.984 in) of water reflection around vessels.

5.2.1.3.2 Enrichment Assumption

The NEF will operate with a 5.0 $^w/_o$ ^{235}U enrichment limit. However, the nuclear criticality safety calculations used an enrichment of 6.0 $^w/_o$ ^{235}U. This assumption provides additional conservatism for plant design.

5.2.1.3.3 Uranium Accumulation and Moderation Assumption

Most components that form part of the centrifuge plant or are connected to it assume that any accumulation of uranium is taken to be in the form of a uranyl fluoride/water mixture at a maximum H/U atomic ratio of 7 (exceptions are discussed in the associated nuclear criticality safety analyses documentation). The ratio is based on the assumption that significant quantities of moderated uranium could only accumulate by reaction between UF_6 and moisture in air leaking into the plant. Due to the high vacuum requirements of a centrifuge plant, in-leakage is controlled at very low levels and thus the H/U ratio of 7 represents an abnormal condition. The maximum H/U ratio of 7 for the uranyl fluoride-water mixture is derived as follows:

The stoichiometric reaction between UF_6 and water vapor in the presence of excess UF_6 can be represented by the equation:

$$UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF$$

Due to its hygroscopic nature, the resulting uranyl fluoride is likely to form a hydrate compound. Experimental studies (Lychev, 1990) suggest that solid hydrates of compositions $UO_2F_2 \cdot 1.5H_2O$ and $UO_2F_2 \cdot 2H_2O$ can form in the presence of water vapor, the former composition being the stable form on exposure to atmosphere.

It is assumed that the hydrate $UO_2F_2 \cdot 1.5H_2O$ is formed and, additionally, that the hydrogen fluoride (HF) produced by the UF_6/water vapor reaction is also retained in the uranic breakdown to give an overall reaction represented by:

$$UF_6 + 3.5H_2O \rightarrow UO_2F_2 \cdot 4HF \cdot 1.5H_2O$$

For the MONK8A (SA, 2001) calculations, the composition of the breakdown product was simplified to $UO_2F_2 \cdot 3.5H_2O$ that gives the same H/U ratio of 7 as above.

In the case of oils, UF_6 pumps and vacuum pumps use a fully fluorinated perfluorinated polyether (PFPE) type lubricant, often referred to by the trade name "Fomblin." Mixtures of UF_6 and PFPE oil would be a less conservative case than a uranyl fluoride/water mixture, since the maximum HF solubility in PFPE is only about 0.1 $^w/_o$. Therefore, the uranyl fluoride/water mixture assumption provides additional conservatism in this case.

5.2.1.3.4 Vessel Movement Assumption

The interaction controls placed on movement of vessels containing enriched uranium are specified in the facility procedures. In general, any item in movement (an item being either an individual vessel or a specified batch of vessels) must be maintained at 60 cm (23.6 in) edge separation from any other enriched uranium, and that only one item of each type, e.g., one trap and one pump, may be in movement at one time. These spacing restrictions are relaxed for vessels being removed from fixed positions. In this situation, one vessel may approach an adjacent fixed plant vessel/component without spacing restrictions.

5.2.1.3.5 Pump Free Volume Assumption

There are two types of pumps used in product and dump systems of the plant:

- The vacuum pumps (product and dump) are rotary vane pumps. In the enrichment plant fixed equipment, these are assumed to have a free volume of 14 L (3.7 gal) and are modeled as a cylinder in MONK8A (SA, 2001). This adequately covers all models likely to be purchased.
- The UF_6 pumping units are a combination unit of two pumps, one 500 m^3/hr (17,656 ft^3/hr) pump with a free volume of 8.52 L (2.25 gal) modeled as a cylinder, and a larger 2000 m^3/hr (70,626 ft^3/hr) pump which is modeled explicitly according to manufacturer's drawings.

5.2.1.4 Nuclear Criticality Safety Analyses

Nuclear criticality safety is analyzed for the design features of the plant system or component and for the operating practices that relate to maintaining criticality safety. The analysis of individual systems or components and their interaction with other systems or components containing enriched uranium is performed to assure the criticality safety criteria are met. The nuclear criticality safety analyses and the safe values in Table 5.1-1, Safe Values for Uniform Aqueous Solution of Enriched UO_2F_2, provide a basis for the plant design and criticality hazards identification performed as part of the Integrated Safety Analysis.

Each portion of the plant, system, or component that may possibly contain enriched uranium is designed with criticality safety as an objective. Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, shows how the safe values of Table 5.1-1, are applied to the facility design to prevent a nuclear criticality event. The NEF is designed and operated in accordance with the parameters provided in Table 5.1-2. The Integrated Safety Analysis reviewed the facility design and operation and identified Items Relied On For Safety to ensure that criticality does not pose an unacceptable risk.

Where there are significant in-process accumulations of enriched uranium as UF_6 the plant design includes multiple features to minimize the possibilities for breakdown of the moderation control limits. These features eliminate direct ingress of water to product cylinders while in process.

Each NCS analysis includes, as a minimum, the following information.

- A discussion of the scope of the analysis and a description of the system(s)/process(es) being analyzed.
- A discussion of the methodology used in the criticality calculations, which includes the validated computer codes and cross section library used and the k_{eff} limit used (0.95).
- A discussion of assumptions (e.g. reflection, enrichment, uranium accumulation, moderation, movement of vessels, component dimensions) and the details concerning the assumptions applicable to the analysis.
- A discussion on the system(s)/process(es) analyzed and the analysis performed, including a description of the accident or abnormal conditions assumed.
- A discussion of the analysis results, including identification of required limits and controls.

During the design phase of NEF, the NCS analysis is performed by a criticality safety engineer and independently reviewed by a second criticality safety engineer. During the operation of NEF, the NCS analysis is performed by criticality safety engineer, independently reviewed by a second criticality safety engineer and approved by the HS&E Manager. Only qualified criticality safety engineers can perform NCS analyses and associated independent review.

5.2.1.5 Additional Nuclear Criticality Safety Analyses Commitments

The NEF NCS analyses were performed using the above methodologies and assumptions. NCS analyses also meet the following:

- NCS analyses are performed using acceptable methodologies.
- Methods are validated and used only within demonstrated acceptable ranges.
- The analyses adhere to ANSI/ANS-8.1-1998 (ANSI, 1998a) as it relates to methodologies.
- The validation report statement in Regulatory Guide 3.71 (NRC, 1998) is as follows: LES has demonstrated (1) the adequacy of the margin of safety for subcriticality by assuring that the margin is large compared to the uncertainty in the calculated value of $k_{eff,}$ (2) that the calculation of k_{eff} is based on a set of variables whose values lie in a range for which the methodology used to determine k_{eff} has been validated, and (3) that trends in the bias support the extension of the methodology to areas outside the area or areas of applicability.

- A specific reference to (including the date and revision number) and summary description of either a manual or a documented, reviewed, and approved validation report for each methodology are included. Any change in the reference manual or validation report will be reported to the NRC by letter.

- The reference manual and documented reviewed validation report will be kept at the facility.

- The reference manual and validation report are incorporated into the configuration management program.

- The NCS analyses are performed in accordance with the methods specified and incorporated in the configuration management program.

- The NCS methodologies and technical practices in NUREG-1520 (NRC, 2002), Section 5.4.3.4, are used to analyze NCS accident sequences in operations and processes.

- The acceptance criteria in NUREG-1520 (NRC, 2002), Section 3.4, as they relate to: identification of NCS accident sequences, consequences of NCS accident sequences, likelihood of NCS accident sequences, and descriptions of IROFS for NCS accident sequences are met.

- NCS controls and controlled parameters to assure that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety are used.

- As stated in ANSI/ANS-8.1-1998 (ANSI, 1998a), process specifications incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.

- ANSI/ANS-8.7-1998 (ANSI, 1998b), as it relates to the requirements for subcriticality of operations, the margin of subcriticality for safety, and the selection of controls required by 10 CFR 70.61(d) (CFR, 2003b), is used.

- ANSI/ANS-8.10-1983 (ANSI, 1983b), as modified by Regulatory Guide 3.71 (NRC, 1998), as it relates to the determination of consequences of NCS accident sequences, is used.

- If administrative k_{eff} margins for normal and credible abnormal conditions are used, NRC pre-approval of the administrative margins will be sought.

- Subcritical limits for k_{eff} calculations such that: k_{eff} subcritical = 1.0 - bias - margin, where the margin includes adequate allowance for uncertainty in the methodology, data, and bias to assure subcriticality are used.

- Studies to correlate the change in a value of a controlled parameter and its k_{eff} value are performed. The studies include changing the value of one controlled parameter and determining its effect on another controlled parameter and k_{eff}.

- The double contingency principle is met. The double contingency principle is used in determining NCS controls and IROFS.

- The acceptance criteria in NUREG-1520 (NRC, 2002) Section 3.4, as they relate to subcriticality of operations and margin of subcriticality for safety, are met.

5.2.1.6 Nuclear Criticality Safety Evaluations (NCSE)

For any change (i.e., new design or operation, or modification to the facility or to activities of personnel, e.g., site structures, systems, components, computer programs, processes, operating procedures, management measures), that involves or could affect uranium, a NCSE shall be prepared and approved. Prior to implementing the change, it shall be determined that the entire process will be subcritical (with approved margin for safety) under both normal and credible abnormal conditions. If this condition cannot be shown with the NCSE, either a new or revised NCS analysis will be generated that meets the criteria, or the change will not be made.

The NCSE shall determine and explicitly identify the controlled parameters and associated limits upon which NCS depends, assuring that no single inadvertent departure from a procedure could cause an inadvertent nuclear criticality and that the safety basis of the facility will be maintained during the lifetime of the facility. The evaluation ensures that all potentially affected uranic processes are evaluated to determine the effect of the change on the safety basis of the process, including the effect on bounding process assumptions, on the reliability and availability of NCS controls, and on the NCS of connected processes.

The NCSE process involves a review of the proposed change, discussions with the subject matter experts to determine the processes which need to be considered, development of the controls necessary to meet the double contingency principle, and identification of the assumptions and equipment (e.g., physical controls and/or management measures) needed to ensure criticality safety.

Engineering judgment of the criticality safety engineer is used to ascertain the criticality impact of the proposed change. The basis for this judgment is documented with sufficient detail in the NCSE to allow the independent review by a second criticality safety engineer to confirm the conclusions of the judgment of results. Each NCSE includes, as a minimum, the following information.

- A discussion of the scope of the evaluation, a description of the system(s)/process(es) being evaluated, and identification of the applicable nuclear criticality safety analysis.
- A discussion to demonstrate the applicable nuclear criticality safety analysis is bounding for the condition evaluated.
- A discussion of the impact on the facility criticality safety basis, including effect on bounding process assumptions, on reliability and availability NCS controls, and on the nuclear criticality safety of connected system(s)/process(es).
- A discussion of the evaluation results, including (1) identification of assumptions and equipment needed to ensure nuclear criticality safety is maintained and (2) identification of limits and controls necessary to ensure the double contingency principle is maintained.

The NCSE is performed and documented by a criticality safety engineer. Once the NCSE is completed and the independent review by a criticality safety engineer is performed and documented, the HS&E Manager approves the NCSE. Only criticality safety engineers who have successfully met the requirements specified in the qualification procedure can perform NCSEs and associated independent review.

The above process for NCSEs is in accordance with ANSI/ANS-8.19-1996 (ANSI, 1996).

5.2.1.7 Additional Nuclear Criticality Safety Evaluations Commitments

NCSEs also meet the following:

- The NCSEs are performed in accordance with the procedures specified and incorporated in the configuration management program.
- The NCS methodologies and technical practices in NUREG-1520 (NRC, 2002), Sections 5.4.3.4.1(10)(a), (b), (d) and (e), are used to evaluate NCS accident sequences in operations and processes.
- The acceptance criteria in NUREG-1520 (NRC, 2002), Section 3.4, as they relate to: identification of NCS accident sequences, consequences of NCS accident sequences, likelihood of NCS accident sequences, and descriptions of IROFS for NCS accident sequences are met.
- NCS controls and controlled parameters to assure that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety are used.
- The double contingency principle is met. The double contingency principle is used in determining NCS controls and IROFS.
- The acceptance criteria in NUREG-1520 (NRC, 2002) Section 3.4, as they relate to subcriticality of operations and margin of subcriticality for safety, are met.

5.3 CRITICALITY ACCIDENT ALARM SYSTEM (CAAS)

The facility is provided with a Criticality Accident Alarm System (CAAS) as required by 10 CFR 70.24, (CFR, 2003d). Areas where Special Nuclear Material (SNM) is handled, used, or stored in amounts at or above the 10 CFR 70.24 (CFR, 2003d) mass limits are provided with CAAS coverage. Emergency management measures are covered in the facility Emergency Plan.

5.4 REPORTING

The following are NCS Program commitments related to event reporting:

- A program for evaluating the criticality significance of NCS events will be provided and an apparatus will be in place for making the required notification to the NRC Operations Center. Qualified individuals will make the determination of significance of NCS events. The determination of loss or degradation of IROFS or double contingency principle compliance will be made against the license and 10 CFR 70 Appendix A (CFR, 2003f).

- The reporting criteria of 10 CFR 70 Appendix A and the report content requirements of 10 CFR 70.50 (CFR, 2003g) will be incorporated into the facility emergency procedures.

- The necessary report based on whether the IROFS credited were lost, irrespective of whether the safety limits of the associated parameters were actually exceeded will be issued.

- If it cannot be ascertained within one hour of whether the criteria of 10 CFR 70 Appendix A (CFR, 2003f) Paragraph (a) or (b) apply, the event will be treated as a one-hour reportable event.

5.5 REFERENCES

ANSI, 1998a. Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, ANSI/ANS-8.1-1998, American National Standards Institute/American Nuclear Society, 1998.

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CFR, 2003c. Title 10, Code of Federal Regulations, Section 70.64, Requirements for new facilities or new processes at existing facilities, 2003.

CFR, 2003d. Title 10, Code of Federal Regulations, Section 70.24, Criticality accident requirements, 2003.

CFR, 2003e. Title 10, Code of Federal Regulations, Section 70.72, Facility changes and change process, 2003.

CFR, 2003f. Title 10, Code of Federal Regulations, Part 70, Appendix A, Reportable Safety Events, 2003.

CFR, 2003g. Title 10, Code of Federal Regulations, Section 70.50, Reporting requirements, 2003.

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TABLES

Table 5.1-1 Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2

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Parameter	**Critical Value k_{eff} = 1.0**	**Safe Value k_{eff} = 0.95**	**Safety Factor**
Values for 5.0 w/o enrichment			
Volume	28.9 L (7.6 gal)	21.6 L (5.7 gal)	0.75
Cylinder Diameter	26.2 cm(10.3 in)	23.6 cm (9.3 in)	0.90
Slab Thickness	12.6 cm (5.0 in)	10.7 cm (4.2 in)	0.85
Water Mass	17.3 kg H_2O (38.1 lb H_2O)	12.7 kg H_2O (28.0 lb H_2O)	0.73
Areal Density	11.9 g/cm^2 (24.4 lb/ft^2)	9.8 g/cm^2 (20.1 lb/ft^2)	0.82
Uranium Mass	37 kg U (81.6 lb U)		
- no double batching		26.6 kg U (58.6 lb U)	0.72
- double batching		16.6 kg U (36.6 lb U)	0.45
Values for 6.0 w/o enrichment			
Volume	24 L (6.3 gal)	18 L (4.8 gal)	0.75
Cylinder Diameter	24.4 cm (9.6 in)	21.9 cm (8.6 in)	0.90
Slab Thickness	11.5 cm (4.5 in)	9.9 cm (3.9 in)	0.86
Water Mass	15.4 kg H_2O (34.0 lb H_2O)	11.5 kg H_2O (25.4 lb H_2O)	0.75
Areal Density	9.5 g/cm^2 (19.5 lb/ft^2)	7.5 g/cm^2 (15.4 lb/ft^2)	0.79
Uranium Mass	27 kg U (59.5 lb U)		
- no double batching		19.5 kg U (43.0 lb U)	0.72
- double batching		12.2 kg U (26.9 lb U)	0.45

Table 5.1-2 Safety Criteria for Buildings/Systems/Components
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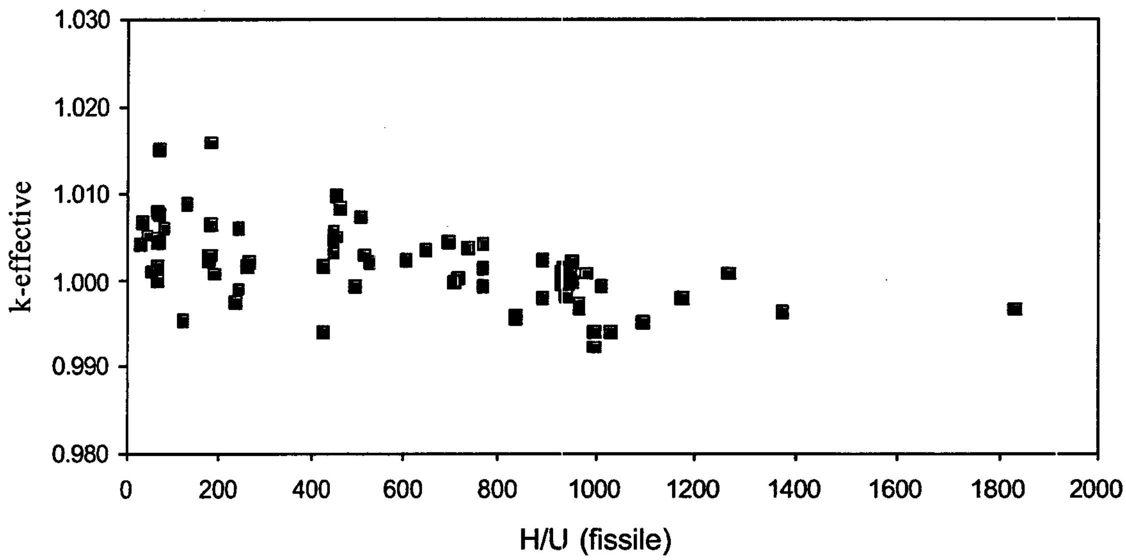
Building/System/Component	Control Mechanism	Safety Criteria
Enrichment	Enrichment	5.0 $^{w}/_{o}$ (6 $^{w}/_{o}$ ^{235}U used in NCS)
Centrifuges	Diameter	< 21.9 cm (8.6 in)
Product Cylinders (30B)	Moderation	H < 0.95 kg (2.09 lb)
Product Cylinders (48Y)	Moderation	H < 1.05 kg (2.31 lb)
UF_6 Piping	Diameter	< 21.9 cm (8.6 in)
Chemical Traps	Diameter	< 21.9 cm (8.6 in)
Product Cold Trap	Diameter	< 21.9 cm (8.6 in)
Contingency Dump System Traps	Enrichment	1.5 $^{w}/_{o}$ ^{235}U
Tanks	Mass	< 12.2 kg U (26.9 lb U)
Feed Cylinders	Enrichment	< 0.72 $^{w}/_{o}$ ^{235}U
Uranium Byproduct Cylinders	Enrichment	< 0.72 $^{w}/_{o}$ ^{235}U
UF_6 Pumps (first stage)	N/A	Safe by explicit calculation
UF_6 Pumps (second stage)	Volume	< 18.0 L (4.8 gal)
Individual Uranic Liquid Containers, e.g., Fomblin Oil Bottle, Laboratory Flask, Mop Bucket	Volume	< 18.0 L (4.8 gal)
Vacuum Cleaners Oil Containers	Volume	<18.0 L (4.8 gal)

Table 5.2-1 Uranium Solution Experiments Used for Validation

MONK8A Case	Case Description	Number of Experiments	Handbook Reference
13	High-enriched uranyl nitrate solutions at various H:U ratios (93.17 $^{w}/_{o}$ ^{235}U)	12	HEU-SOL-THERM-002 HEU-SOL-THERM-003
23	Uranyl nitrate solution (~ 95 $^{w}/_{o}$ enriched)	5	HEU-SOL-THERM-013 NS&E
35	High-enriched uranyl nitrate solutions (U concentration from 20-700 g/L)	11	HEU-SOL-THERM-009 - HEU-SOL-THERM-012
43	Low-enriched uranyl nitrate solutions	3	LEU-SOL-THERM-002
51	Low-enriched uranium solutions (new STACY experiments)	7	LEU-SOL-THERM-004
63	Boron carbide absorber rods in uranyl nitrate (5.6 $^{w}/_{o}$ enriched)	3	LEU-SOL-THERM-005
67	Highly enriched uranyl nitrate solution with a concentration range between 59.65 and 334.66 g U/L	10	HEU-SOL-THERM-001
68	Highly enriched uranyl fluoride/heavy water solution with a concentration range between 60 and 679 g U/L and a heavy water reflector	6	HEU-SOL-THERM-004
71	STACY: 28 cm thick slabs of 10 $^{w}/_{o}$ enriched uranyl nitrate solutions, water reflected	7	LEU-SOL-THERM-016
80	STACY: Unreflected 10 $^{w}/_{o}$ enriched uranyl nitrate solution in a 60 cm diameter cylindrical tank	5	LEU-SOL-THERM-007
81	STACY: Concrete reflected 10 $^{w}/_{o}$ enriched uranyl nitrate solution reflected by concrete	4	LEU-SOL-THERM-008
84	STACY: Borated concrete reflected 10 $^{w}/_{o}$ enriched uranyl nitrate solution in a 60 cm diameter cylindrical tank	3	LEU-SOL-THERM-009
85	STACY: Polyethylene reflected 10 $^{w}/_{o}$ enriched uranyl nitrate solution in a 60 cm diameter cylindrical tank	4	LEU-SOL-THERM-010

FIGURES



REFERENCE NUMBER
Figure 5.2–1.doc

FIGURE 5.2-1
VALIDATION RESULTS FOR
URANIUM SOLUTIONS

REVISION DATE: DECEMBER 2003

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6.0 CHEMICAL PROCESS SAFETY

This chapter describes the Louisiana Energy Services (LES) plan for managing chemical process safety and demonstrating that chemical process safety controls meet the requirements of 10 CFR 70 (CFR, 2003a) thereby providing reasonable assurance that the health and safety of the public and facility employees is protected. The chapter describes the chemical classification process, the hazards of chemicals of concern, process interactions with chemicals affecting licensed material and/or hazardous chemicals produced from licensed material, the methodology for evaluating hazardous chemical consequences, and the chemical safety assurance features.

The chemical process safety program for the National Enrichment Facility (NEF) is similar to attributes for chemical safety which were submitted for Nuclear Regulatory Commission (NRC) review in the LES license application for the Claiborne Enrichment Center (LES, 1993). The NRC staff evaluated these prior attributes and concluded in NUREG-1491 (NRC, 1994) that the operation of the facility would be adequately safe with respect to chemical processes and hazards.

The NEF chemical process safety program meets the acceptance criteria in Chapter 6 of NUREG-1520 (NRC, 2002) and complies with 10 CFR 70.61 (CFR, 2003b), 70.62 (CFR, 2003c) and 70.64 (CFR, 2003d).

The information provided in this chapter, the corresponding regulatory requirement and the section of NUREG-1520 (NRC, 2002) Chapter 6 in which the NRC acceptance criteria are presented are summarized below:

Information Category and Requirement	10 CFR 70 Citation	NUREG-1520 Chapter 6 Reference
Section 6.1 Chemical Information		
• Properties and Hazards	70.62(c)(1)(ii)	6.4.3.1
Section 6.2 Chemical Process Information		
• General Information	70.65(b)(3)	6.4.3.1
• Design Basis, Materials, Parameters	70.62(b)	6.4.3.1
• Process Chemistry, Chemical Interaction		6.4.3.2
Section 6.3 Chemical Hazards Analysis		
• Methodology, Scenarios, Evaluation	70.65(b)(3)	6.4.3.2
Section 6.4 Chemical Safety Assurance		
• Management, Configuration Control, Design, BDC, Maintenance, Training, Procedures, Audits, Emergency Planning, Incident Investigation	70.65(b)(4)	6.4.3.2 6.4.3.3

6.1 CHEMICAL INFORMATION

This section addresses the criteria utilized to classify all site chemicals based on their potential for harm and as defined by regulatory requirements. It also presents information on the properties of those chemicals.

6.1.1 Chemical Screening and Classification

Table 6.1-1, Chemicals – Hazardous Properties, provides the listing of chemicals and related chemical wastes that are expected to be in use at the NEF. Chemical formulas in this Chapter utilize subscripting per standard convention. The hazardous properties of each chemical and related chemical waste have been listed. Also, each chemical or related waste has been classified into one of three categories (NEF Classes): Chemicals of Concern (Class 1), Interaction Chemicals (Class 2), or Incidental Chemicals (Class 3).

The definition of each classification is provided below.

Tables 6.1-2 through 6.1-5 are the basic chemical inventories for the facility. Each of these tables lists a major facility structure, area, and/or system and an associated inventory of significant chemicals/chemical usage for each area. These tables do not include the listing of all incidental sludges, wastes, and waste streams which are presented in Table 6.1-1 and do not include those chemicals that have been characterized as Class 3 materials and that are not a stored "chemical". As such, those chemicals not included are not a process safety concern. Complete inventories of chemicals and chemical wastes (including incidental sludges, wastes, and waste streams) by area are provided in Chapter 2 of the Environmental Report.

6.1.1.1 Chemicals of Concern (Class 1)

Chemicals of Concern (NEF Class 1) are determined based on one or more characteristics of the chemical and/or the quantity in storage/use at the facility. For licensed material or hazardous chemicals produced from licensed materials, chemicals of concern are those that, in the event of release have the potential to exceed any of the concentrations defined in 10 CFR 70 (CFR, 2003a) as listed below.

<u>High Risk Chemicals of Concern</u>

1. An acute worker dose of 1 Sv (100 rem) or greater total effective dose equivalent.
2. An acute dose of 0.25 Sv (25 rem) or greater total effective dose equivalent to any individual located outside the controlled area.
3. An intake of 30 mg or greater of uranium in soluble form by any individual located outside the controlled area.

4. An acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that:
   (i) Could endanger the life of a worker, or
   (ii) Could lead to irreversible or other serious, long-lasting health effects to any individual located outside the controlled area.

<u>Intermediate Risk Chemicals of Concern</u>

1. An acute worker dose of 0.25 Sv (25 rem) or greater total effective dose equivalent.
2. An acute dose of 0.05 Sv (5 rem) or greater total effective dose equivalent to any individual located outside the controlled area.
3. A 24-hour averaged release of radioactive material outside the restricted area in concentrations exceeding 5000 times the values in Table 2 of Appendix B to 10 CFR 20 (CFR, 2003e).
4. An acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that:
   (i) Could lead to irreversible or other serious, long-lasting health effects to a worker, or
   (ii) Could cause mild transient health effects to any individual located outside the controlled area.

<u>Non-Licensed Chemicals of Concern</u>

For those chemicals that are not related to licensed materials, chemicals of concern are those that are listed and handled above threshold quantities of either of the following standards:

1. 29 CFR 1910.119 (CFR, 2003f) – OSHA Process Safety Management
2. 40 CFR, 68 (CFR, 2003g) – EPA Risk Management Program.

These chemicals represent, based on their inherent toxic, reactive, or flammable properties, a potential for severe chemical release and/or acute chemical exposure to an individual that:

(i) Could endanger the life of a worker, or

(ii) Could lead to irreversible or other serious, long-lasting health effects to any individual located outside the controlled area.

It is noted here, that uranium hexafluoride (UF_6) is the only licensed material-related chemical of concern (NEF Class 1) that will be used at the facility. There are no non-licensed chemicals of concern at the facility.

6.1.1.2 Interaction Chemicals (Class 2)

Interaction chemicals (NEF Class 2) are those chemicals/chemical systems that require evaluation for their potential to precipitate or propagate accidents in chemical of concern (NEF Class 1) systems, but by themselves are not chemicals of concern.

6.1.1.3 Incidental Chemicals (Class 3)

The facility will use other chemicals that are neither chemicals of concern nor interaction chemicals. Some of these incidental chemicals (NEF Class 3) include those that have the potential to result in injurious occupational and/or environmental exposure, but represent no potential for acute exposure to the public and which via their nature, quantity, and/or use, have no potential for impacting chemicals of concern (NEF Class 1).

These chemicals will not be subject to chemical process safety controls. Controls will be placed on incidental chemical storage, use and handling as necessary and as follows:

1. General occupational chemical safety controls will be in place for protection of facility employees in the storage, handling, and use of all chemicals as required by 29 CFR 1910 (CFR, 2003h)

2. Environmental protection controls required to prevent and/or mitigate environmental damage due to spills and discharges and to control anticipated effluents and waste are detailed in Chapter 9, Environmental Protection, and the NEF Environmental Report.

6.1.2 Chemicals of Concern - Properties

This section summarizes the chemical properties for chemicals of concern and their key byproducts.

6.1.2.1 Uranium Hexafluoride - Chemical Properties

6.1.2.1.1 Physical

Uranium hexafluoride (UF_6) is a chemical compound consisting of one atom of uranium combined with six atoms of fluorine. It is the chemical form of uranium that is used during the uranium enrichment process.

UF_6 can be a solid, liquid, or gas, depending on its temperature and pressure. Multiple phases coexist in equilibrium only under exact combinations of temperature and pressure. These properties are shown in Figure 6.1-1, UF_6 Phase Diagram, which presents the different physical forms of UF_6 as a function of temperature and pressure. The three phases are identified as regions on the diagram separated by lines representing a plot of equilibrium combinations of temperature and pressure. These boundaries all converge at one unique point on the diagram, called the triple point, where all three phases coexist in equilibrium. The triple point of UF_6 is 64°C (147°F) and 152 kPa (22 psia).

Liquid UF_6 is formed only at temperatures and pressures greater than the triple point. Below the triple point, solid UF_6 will change phase directly to UF_6 gas (sublimation) when the temperature is raised and/or the pressure is lowered at continuous points along the solid/gas interface line. This will occur without the UF_6 progressing through a liquid phase. Solid UF_6 is a white, dense, crystalline material that resembles rock salt. Both liquid and gaseous UF_6 are colorless.

Pure UF_6 follows its phase diagram consistently regardless of isotopic content. Impurities in a UF_6 cylinder will cause deviations in the normal phase behavior. The most common gaseous impurities in UF_6 feed are air and hydrogen fluoride (HF) which are generated from the reaction of UF_6 with moisture in the air. Since these light gas impurities have a higher vapor pressure than UF_6, their presence can be detected by measuring the static pressure of cylinders and comparing the results to the UF_6 phase diagram (when the UF_6 temperature is known).

UF_6 exhibits significant expansion when going from solid to liquid phase and continues to expand as the liquid temperature increases. This is illustrated in Figure 6.1-2, Densities of Solid and Liquid UF_6. This figure shows that UF_6 expands roughly 53% going from a solid at 21°C (70°F) to a liquid at 113°C (235°F). Department of Transportation cylinder fill limits are based on UF_6 density at 121°C (250°F) and provide five percent ullage or free volume as a safety factor to prevent hydraulic rupture due to heating.

Other physical properties of UF_6 are presented in Table 6.1-6, Physical Properties of UF_6.

6.1.2.1.2 Reactivity

UF_6 does not react with oxygen, nitrogen, carbon dioxide, or dry air, but it does react with water. For this reason, UF_6 is handled in leak tight containers and processing equipment. When UF_6 comes into contact with water, such as the water vapor in the air, the UF_6 and water react, forming hydrogen fluoride (HF) gas and a solid uranium-oxyfluoride compound (UO_2F_2) which is commonly referred to as uranyl fluoride. Additional information on UF_6 reactions with water is provided in Section 6.2.1, Chemistry and Chemical Reactions.

UF_6 is also incompatible with a number of other chemicals including hydrocarbons and aromatics but none of these chemicals are used in or within proximity of UF_6 process systems.

6.1.2.1.3 Toxicological

If UF_6 is released to the atmosphere, the uranium compounds and HF that are formed by reaction with moisture in the air are chemically toxic. Uranium is a heavy metal that, in addition to being radioactive, can have toxic chemical effects (primarily on the kidneys) if it enters the bloodstream by means of ingestion or inhalation. HF is an extremely corrosive gas that can damage the lungs and cause death if inhaled at high enough concentrations. Additional information on the toxicological parameters used for evaluating exposure is provided in Section 6.3, Chemical Hazards Analysis.

6.1.2.1.4 Flammability

UF_6 is not flammable and does not disassociate to flammable constituents under conditions at which it will be handled at the facility.

6.1.2.2 Hydrogen Fluoride - Chemical Properties

Hydrogen fluoride (HF) is not a direct chemical of concern (NEF Class 1), however, it is one of two byproducts of concern that would be developed in the event of most accident scenarios at the facility. Understanding its properties therefore is important in evaluating chemical process conditions.

6.1.2.2.1 Physical

HF can exist as a gas or as a liquid under pressure (anhydrous hydrogen fluoride) or as an aqueous solution of varying strengths (aqueous hydrofluoric acid). HF vapors are colorless with a pungent odor which is detectable at concentrations above 1 ppm. It is soluble in water with a release of heat.

Releases of anhydrous hydrogen fluoride would typically fume (due to the reaction with water vapor) so that any significant release would be visible at the point of release and in the immediate vicinity.

6.1.2.2.2 Reactivity

In both gaseous and aqueous form, HF is extremely reactive, attacking certain metals, glass and other silicon-containing components, leather and natural rubber. Additional information regarding the corrosion properties and metal attack are provided in Section 6.2.1.3, UF_6 and Construction Materials.

6.1.2.2.3 Toxicological

HF in both gaseous and aqueous forms is strongly corrosive and causes severe burns to the skin, eyes and mucous membranes and severe respiratory irritation.

Inhalation of HF causes an intolerable prickling, burning sensation in the nose and throat, with cough and pain beneath the sternum. Nausea, vomiting, diarrhea and ulceration of the gums may also occur. In low concentrations, irritation of the nasal passages, dryness, bleeding from the nose and sinus disorders may result, while continued exposure can lead to ulceration and perforation of the nasal septum. Exposure to high concentrations can cause laryngitis, bronchitis and pulmonary edema which may not become apparent until 12-24 hours after the exposure.

Chronic exposure to excessive quantities of gaseous or particulate fluoride results in nausea, vomiting, loss of appetite and diarrhea or constipation. Fluorosis and other chronic effects may result from significant acute exposures. Systemic fluoride poisoning can cause hypocalcaemia which may lead to cardiac arrhythmias and/or renal failure. Chronic exposure to gaseous or particulate fluoride is not expected at the facility.

Skin exposure to concentrated liquid HF will result in aggressive chemical burns. Burns from exposure to dilute solutions (1-20%) of hydrofluoric acid (aqueous HF) or moderate concentrations of vapor may not be immediately painful or visible. Symptoms of skin exposure include immediate or delayed throbbing, burning pain followed by localized destruction of tissue and blood vessels that may penetrate to the bone. Exposure to liquid forms of HF is not expected at the facility.

Ocular exposure to HF causes a burning sensation, redness and secretion. Splashes of aqueous hydrofluoric acid to the eye rapidly produce conjunctivitis, keratitis and more serious destructive effects but these are not expected at the facility.

6.1.2.2.4 Flammability

HF is not flammable or combustible. HF can react exothermically with water to generate sufficient heat to ignite nearby combustibles. HF in reaction with certain metals can offgas hydrogen which is flammable. Both of these reactions would be more typical for bulk, concentrated HF interaction where large masses (i.e., bulk HF storage) of material are involved. These types of interactions are not expected at the facility.

6.1.2.3 Uranyl Fluoride - Chemical Properties

Uranyl fluoride (UO_2F_2) is not a direct chemical of concern (NEF Class 1), however, it is the second of two byproducts of concern (HF is the other) that would be developed in the event of a UF_6 release at the facility. Understanding its properties therefore is important in evaluating chemical process conditions.

6.1.2.3.1 Physical

UO_2F_2 is an intermediate in the conversion of UF_6 to a uranium oxide or metal form and is a direct product of the reaction of UF_6 with moisture in the air. It exists as a yellow, hygroscopic solid. UO_2F_2 formation and dispersion is governed by the conditions of the atmosphere in which the release is occurring. UF_6 will be continually hydrolyzed in the presence of water vapor. The resulting UF_6/HF cloud will include UO_2F_2 particulate matter within the gaseous stream. As this stream diffuses into larger volumes and additional UF_6 hydrolysis occurs, UO_2F_2 particulate will settle on surfaces as a solid flake-like compound. This deposition will occur within piping/equipment, on lower surfaces within enclosures/rooms, and/or on the ground – wherever the UF_6 hydrolysis reaction is occurring.

6.1.2.3.2 Reactivity

UO_2F_2 is reported to be stable in air to 300°C (570°F). It does not have a melting point because it undergoes thermal decomposition to triuranium octoxide (U_3O_8) above this temperature. When heated to decomposition, UO_2F_2 emits toxic fluoride fumes. UO_2F_2 is hygroscopic and water-soluble and will change in color from brilliant orange to yellow after reacting with water.

6.1.2.3.3 Toxicological

UO_2F_2 is radiologically and chemically toxic due to its uranium content and solubility. Once inhaled, uranyl fluoride is easily absorbed into the bloodstream because of its solubility. If large quantities are inhaled, the uranium in the uranyl complex acts as a heavy metal poison that affects the kidneys. Because of low specific activity values, the radiological toxicity of UF_6 and the UO_2F_2 byproduct are typically of less concern than the chemical toxicity.

6.1.2.3.4 Flammability

UO_2F_2 is not combustible and will not decompose to combustible constituents under conditions at which it will be handled at the facility.

6.2 CHEMICAL PROCESS INFORMATION

This section characterizes chemical reactions between chemicals of concern and interaction chemicals and other substances as applicable. This section also provides a basic discussion of the chemical processes associated with UF_6 process systems.

6.2.1 Chemistry and Chemical Reactions

Although the separation of isotopes is a physical rather than chemical process, chemical principles play an important role in the design of the facility. The phase behavior of UF_6 is critical to the design of all aspects of the plant. UF_6 has a high affinity for water and will react exothermically with water and water vapor in the air. The products of UF_6 hydrolysis, solid UO_2F_2 and gaseous HF, are both toxic. HF is also corrosive, particularly in the presence of water vapor. Because this chemical reaction results in undesirable by-products, UF_6 is isolated from moisture in the air through proper design of primary containment (i.e., piping, components, and cylinders).

Other chemical reactions occur in systems that decontaminate equipment, remove contaminants from effluent streams, and as part of lubricant recovery or other cleansing processes. Side reactions can include the corrosion and deterioration of construction materials, which influences their specification. These reactions are further described below.

6.2.1.1 UF_6 and Water

Liquid and gaseous UF_6 react rapidly with water and water vapor as does the exposed surface of solid UF_6. UF_6 reacts with water so rapidly that the HF formed is always anhydrous when in the presence of UF_6, significantly reducing its corrosive potential in cylinders, piping and equipment. The reaction of gaseous UF_6 with water vapor at elevated temperatures is shown in Equation 6.2-1.

$$\underset{(gas)}{UF_6} + \underset{(vapor)}{2\,H_2O} \Rightarrow \underset{(solid)}{UO_2F_2} + \underset{(gas)}{4HF} + heat \qquad \text{(Eq. 6.2-1)}$$

At room temperature, depending on the relative humidity of the air, the products of this reaction are UO_2F_2 hydrates and HF- H_2O fog, which will be seen as a white cloud. A typical reaction with excess water is given in Equation 6.2-2.

$$\underset{(gas)}{UF_6} + \underset{(vapor)}{(2+4x)H_2O} \Rightarrow \underset{(solid)}{UO_2F_2 * 2\,H_2O} + \underset{(fog)}{4HF * x\,H_2O} + heat \qquad \text{(Eq. 6.2-2)}$$

If, because of extremely low humidity, the HF- H_2O fog is not formed, the finely divided uranyl fluoride (UO_2F_2) causes only a faint haze. UO_2F_2 is a water-soluble, yellow solid whose exact coloring depends on the degree of hydration as well as the particle size.

The heat release for the reaction in Equation 1 is 288.4 kJ/kg (124 BTU/lbm) of UF_6 gas reacted. The heat release is much larger if the UO_2F_2 is hydrated and HF-H_2O fog is formed with a heat release of 2,459 kJ/kg (1057 BTU/lbm) of UF_6 vapor.

These reactions, if occurring in the gaseous phase at ambient or higher temperatures, are very rapid, near instantaneous. Continuing reactions between solid UF_6 and excess water vapor occur more slowly as a uranyl fluoride layer will form on surface of the solid UF_6 which inhibits the rate of chemical reaction.

UF_6 reactions with interaction chemicals are discussed below. These include chemical reactions associated with lubricants and other chemicals directly exposed to UF_6, as well as chemicals used to recover contaminants from used lubricating oils, and capture trace UF_6, uranium compounds, and HF from effluent streams. UF_6 reactions with materials of construction are addressed in Section 6.2.1.3, UF_6 and Construction Material.

6.2.1.2 UF_6 and Interaction Chemicals

The chemistry of UF_6 is significantly affected by its fluorination and oxidation potential. Many of the chemical properties of UF_6 are attributable to the stability of the UO_2^{++} ion, which permits reactions with water, oxides, and salts containing oxygen-bearing anions such as SO_4^{--}, NO_3^{--}, and CO_3^{--} without liberation of the O_2 molecule.

The following subsection describes potential chemical interactions between the UF_6 process streams and interaction chemicals.

6.2.1.2.1 PFPE (Fomblin) Oil

The reaction of UF_6 with hydrocarbons is undesirable and can be violent. Gaseous UF_6 reacts with hydrocarbons to form a black residue of uranium-carbon compounds. Hydrocarbons can be explosively oxidized if they are mixed with UF_6 in the liquid phase or at elevated temperatures. *It is for this reason that non-fluorinated hydrocarbon lubricants are not utilized in* any UF_6 system at the NEF.

UF_6 vacuum pumps are lubricated using PFPE (Perfluorinated Polyether) oil which is commonly referred to by a manufacturer's trade name - Fomblin oil. Fomblin oil is inert, fully fluorinated and does not react with UF_6 under any operating conditions.

Small quantities of uranium compounds and traces of hydrocarbons may be contained in the Fomblin oil used in the UF_6 vacuum pumping systems. The UF_6 degrades in the oil or reacts with trace hydrocarbons to form crystalline compounds – primarily uranyl fluoride (UO_2F_2) and uranium tetrafluoride (UF_4) particles – that gradually thicken the oil and reduce pump capacity.

Recovery of Fomblin oil for reuse in the system is conducted remotely from the UF_6 process systems. The dissolved uranium compounds are removed in a process of precipitation, centrifugation, and filtration. Anhydrous sodium carbonate (Na_2CO_3) is added to contaminated

Fomblin oil. Uranium compounds react to form sodium uranyl carbonate, which precipitates out. A filter removes the precipitate during subsequent centrifugation of the oil.

Trace amounts of hydrocarbons are then removed by adding activated carbon to the Fomblin oil and heating causing absorption of the hydrocarbons. The carbon is in turn removed through a bed of celite.

Failures associated with Fomblin oil and Fomblin oil recovery were evaluated in the Integrated Safety Analysis.

6.2.1.2.2 Chemical Traps - Activated Carbon, Aluminum Oxide, and Sodium Fluoride

Adsorption is the attraction of gas molecules to the surface of an activated solid. There are two classifications of adsorption: physical and chemical. At ordinary temperatures, adsorption is usually caused by molecular forces rather than by the formation of chemical bonds. In this type of adsorption, called physical adsorption, very little heat is evolved. If a chemical reaction takes place between the gas and the solid surface, the process is known as chemisorption. In chemisorption the reaction between surface and gas molecules occurs in a stoichiometric manner, and heat is liberated during the reaction.

Chemisorption is used in the removal of UF_6 and HF from gaseous effluent streams. It is also used to remove oil mist from vacuum pumps operating upstream of gaseous effluent ventilation systems. Adsorbent materials are placed on stationary beds in chemical traps downstream of the various cold traps. These materials capture HF and the trace amounts of UF_6 that escape desublimation during feed purification or during venting of residual UF_6 contained in hoses and/or piping that is bled down before disconnection.

The chemical traps are placed in series downstream of the cold traps in the exhaust streams to the Gaseous Effluent Vent Systems (GEVS) and may include one or more of a series of two different types of chemical traps. The first type of trap contains a charge of activated carbon to capture the small amounts of UF_6 that escape desublimation. Since chemisorption is a pressure sensitive process, HF is not fully adsorbed on carbon at low pressures. This necessitates a second type of trap containing a charge of aluminum oxide (Al_2O_3) to remove HF from the gaseous effluent stream. One or more of a series of these traps is used depending on the process system being served. Additionally, a carbon trap is present on the inlet of the vacuum pumps which discharge to the GEVS to prevent any of the pump oil from migrating back into the UF_6 cold traps.

Chemisorption of UF_6 on activated carbon evolves considerable thermal energy. This is not normally a problem in the chemical traps downstream of the cold traps because very little UF_6 escapes desublimation. If multiple equipment failures and/or operator errors occur, significant quantities of UF_6 could enter the chemical traps containing activated carbon. This could cause significant overheating leading to release. Failures associated with the carbon traps were evaluated in the Integrated Safety Analysis.

Activated carbon cannot be used in the Contingency Dump System because the relatively high UF_6 flow rates during this non-routine operation could lead to severe overheating. A chemical trap containing sodium fluoride (NaF) is installed in the contingency dump flow path to trap UF_6. NaF is used because the heat of UF_6 chemisorption on NaF is significantly lower than the heat of UF_6 chemisorption on activated carbon. Failures associated with the NaF traps were evaluated in the integrated safety analysis.

There are no specific concerns with heat of adsorption of either UF_6 or HF with Al_2O_3. Failures associated with the aluminum oxide traps were evaluated in the Integrated Safety Analysis.

The properties of these chemical adsorbents are provided in Table 6.2-1, Properties of Chemical Adsorbents.

6.2.1.2.3 Decontamination – Citric Acid

Contaminated components (e.g., pumps, valves, piping), once they are removed from the process areas, undergo decontamination. Oily parts are washed in a hot water wash that will remove the bulk of oil including residual uranic compounds. Once the hot water wash is complete, citric acid is used to remove residual uranic fluoride compound layers that are present on the component surfaces. The reaction of the uranium compounds with the citric acid solution produces various uranyl citrate complexes. After citric acid cleansing, the decontaminated component is subject to two additional water wash/rinse cycles. The entire decontamination operation is conducted in small batches on individual components.

Decontamination of sample bottles and valves is also accomplished using citric acid.

Decontamination was evaluated in the Integrated Safety Analysis. Adequate personnel protective features are in place for safely handling decontamination chemicals and byproducts.

6.2.1.2.4 Nitrogen

Gaseous nitrogen is used in the UF_6 systems for purging and filling lines that have been exposed to atmosphere for any of several reasons including: connection and disconnection of cylinders, preparing lines/components for maintenance, providing an air-excluding gaseous inventory for system vacuum pumps, and filling the interstitial space of the liquid sampling autoclave (secondary containment) prior to cylinder liquefaction.

The nitrogen system consists of a liquid nitrogen bulk storage vessel, vaporizer, gaseous nitrogen heater, liquid and gaseous nitrogen distribution lines and instrumentation. Liquid nitrogen is delivered by tanker and stored in the storage vessel.

Nitrogen is not reactive with UF_6 in any plant operational condition. Failures of the nitrogen system were evaluated in the Integrated Safety Analysis.

6.2.1.2.5 Silicone Oil

Silicone oil is used as a heat exchange medium for the heating/chilling of various cold traps. This oil is external to the UF_6 process stream in all cases and is not expected to interact with UF_6. Failures in the heating/chilling systems were evaluated in the Integrated Safety Analysis.

6.2.1.2.6 Halocarbon Refrigerants

Halocarbon refrigerants (including R23 trifluoromethane, R404A fluoromethane blend, and R507 penta/trifluoromethane) are used in individual package chillers that will provide cooling of UF_6 cylinders and/or silicon oil heat exchange media for take-off stations and cold traps. These halocarbons were selected due to good heat transfer properties, because they satisfy environmental restrictions regarding ozone depletion, and are non-flammable. All halocarbon refrigerants are external to the UF_6 process stream in all cases and are not expected to interact with UF_6. Failures in the heating/chilling systems were evaluated in the Integrated Safety Analysis.

6.2.1.2.7 Plant Chilled Water

Chilled water is circulated in coils as a heat exchange medium for cooling of the liquid sampling autoclave after liquid samples have been drawn. Chilled water is external to the autoclave which is secondary containment for the product cylinder and sampling piping representing three physical barriers between the water and the UF_6 so no interaction is anticipated. Failures in the chilled water distribution system were evaluated in the Integrated Safety Analysis.

6.2.1.2.8 Centrifuge Cooling Water

Centrifuge cooling water is provided from the Centrifuge Cooling Water Distribution System. The function of this system is to provide a supply of deionized cooling water to the cooling coils of the centrifuges. This system provides stringent control over the operating temperature of the centrifuges to enable their efficient operation. Centrifuge cooling water is external to the UF_6 process stream in all cases and is not expected to interact with UF_6. Failures in the centrifuge cooling water distribution system were evaluated in the Integrated Safety Analysis.

6.2.1.3 UF_6 and Construction Materials

The corrosion of metallic plant components and the deterioration of non-metallic sealing materials is avoided by specifying resistant materials of construction and by controlling process fluid purity.

Direct chemical attack by the process fluid on metallic components is the result of chemical reactions. In many cases, the affinity of the process fluid for the metal produces metallic compounds, suggesting that rapid destruction of the metal would take place. This is usually prevented by the formation of a protective layer on the surface of the metal.

Deterioration of non-metallic materials is caused by exposure to process fluids and conditions. Materials used in gaskets, valves, flexible hoses, and other sealants must be sufficiently inert to have a useful service life.

UF_6 and some of its reaction products are potentially corrosive substances, particularly HF. UF_6 is a fluorinating agent that reacts with most metals. The reaction between UF_6 and metals such as nickel, copper, and aluminum produces a protective fluoride film over the metal that inhibits further reaction. These materials are therefore relatively inert to UF_6 corrosion after passivation and are suitable for UF_6 service. Aluminum is used as piping material for UF_6 systems because it is especially resistant to corrosion in the presence of UF_6. Carbon steels and stainless steels can be attacked by UF_6 at elevated temperatures but are not significantly affected by the presence of UF_6 at the operating temperatures for the facility.

Light gas impurities such as HF and air are removed from UF_6 during the purification process. Although HF is a highly corrosive substance when in solution with water as aqueous hydrofluoric acid, it contributes very little to metal corrosion when in the presence of UF_6. This is due to the fact that UF_6 reacts with water so rapidly that HF remains anhydrous when in the presence of UF_6.

Corrosion rates of certain metals in contact with UF_6 are presented in Table 6.2-2, UF_6 Corrosion Rates, for two different temperatures. This data was provided in the original Safety Analysis Report for the Claiborne Enrichment Center (LES, 1993).

Resistant metal such as stainless steel are used in valve bellows and flex hoses. Aluminum piping is bent to minimize the use of fittings. Connections are welded to minimize the use of flanges and gaskets. As a standard practice, the use of sealant materials is minimized to reduce the number of potential leak paths.

Non-metallic materials are required to seal connections in UF_6 systems to facilitate valve and instrument replacement as well as cylinder connections. They are also used in valve packing and seating applications. All gasketing and packing material used at the facility will be confirmed as appropriate for UF_6 services. Typical materials that are resistant to UF_6 through the range of plant operating conditions include butyl rubber, Viton, and Kel-F.

The materials used to contain UF_6 are provided in Table 6.2-3, Materials of Construction for UF_6 Systems. The cylinders to be used at the facility are standard Department of Transportation approved containers for the transport and storage of UF_6, designed and fabricated in accordance with ANSI N14.1 (ANSI, applicable version). The nominal and minimum (for continued service) wall thickness for cylinders listed in Table 6.2-3, are taken from this standard.

The remaining system materials are relatively inert in the presence of UF_6 and the corrosion rates given in Table 6.2-2, indicate that these materials are acceptable for UF_6 service over the life of the plant.

As shown in Table 6.2-3, the cylinders used to store and transport UF_6 are made of carbon steel. Uranium Byproduct Cylinders (UBCs) are stored outside in open air where they are exposed to the elements. Atmospheric corrosion is determined by the exposure to moisture (e.g., rain, snow, atmospheric humidity) and the impurities in the air (such as sulfur). The corrosion rate on the outside surfaces of the carbon steel cylinders therefore varies accordingly with these conditions. Carbon steel storage cylinders are painted to provide a corrosion barrier to external elements.

External corrosion can occur on the outside cylinder surface and at interface points such as the contact point with the resting blocks and in skirt depressions (at the cylinder ends). According to a paper entitled Monitoring of Corrosion in ORGDP Cylinder Yards (DOE, 1988), the average corrosion rate experienced by UBCs is less than 0.051 mm/yr (2 mils/yr). This corrosion rate is almost exclusively due to exterior rust on the carbon steel. Another report – Prediction of External Corrosion for Steel Cylinders – 2001 Report (ORNL, 2001) – sampled exterior steel cylinders (30A) at Oak Ridge National Laboratories that had been subject to intermittent contact with the ground and found to have average corrosion rates of approximately 0.041 mm/yr (1.6 mils/yr). These values indicate that the expected service life would be greater than 50 years. These rates are conservative based on the UBC storage arrangement at the NEF. Cylinders subject to weather conditions (i.e., UBCs) will be periodically inspected to assess corrosion and corrosion rate.

6.2.2 Process - General Enrichment Process

Uranium enrichment is the process by which the isotopic composition of uranium is modified. Natural uranium consists of three isotopes, uranium 234 (^{234}U), uranium 235 (^{235}U), and uranium 238 (^{238}U), approximately 0.0058 $^{w}/_{o}$, 0.711 $^{w}/_{o}$ and 99.28 $^{w}/_{o}$ respectively. ^{235}U, unlike ^{238}U, is fissile and can sustain a nuclear chain reaction. Light water nuclear power plants (the type in the United States) normally operate on fuel containing between 2 $^{w}/_{o}$and 5 $^{w}/_{o}$ ^{235}U (low-enriched uranium); therefore, before natural uranium is used in uranium fuel for light water reactors it undergoes "enrichment."

In performing this enrichment, the NEF will receive and enrich natural uranium hexafluoride (UF_6) feed. The isotopes are separated in gas centrifuges arranged in arrays called cascades.

This process will result in the natural UF_6 being mechanically separated into two streams: (1) a product stream which is selectable up to a maximum 5 $^{w}/_{o}$ ^{235}U enrichment, and (2) a tails stream which is depleted to low percentages of ^{235}U (0.32 $^{w}/_{o}$ on average). No chemical reaction occurs during enrichment. Other processes at the plant include product blending, homogenizing and liquid sampling to ensure compliance with customer requirements and to ensure a quality product.

The enrichment process is comprised of the following major systems:

- UF_6 Feed System
- Cascade System
- Product Take-Off System
- Tails Take-Off System
- Product Blending System
- Product Liquid Sampling System.

UF_6 is delivered to the plant in ANSI N14.1 (ANSI, applicable version) standard Type 48X or 48Y international transit cylinders, which are placed in a feed station and connected to the plant via a common manifold. Heated air is circulated around the cylinder to sublime UF_6 gas from the solid phase. The gas is flow controlled through a pressure control system for distribution to the cascade system at subatmospheric pressure.

Individual centrifuges are not able to produce the desired product and tails concentration in a single step. They are therefore grouped together in series and in parallel to form arrays known as cascades. A typical cascade is comprised of many centrifuges.

UF_6 is drawn through cascades with vacuum pumps and compressed to a higher subatmospheric pressure at which it can desublime in the receiving cylinders. Highly reliable UF_6 resistant pumps will be used for transferring the process gas.

Tails material and product material are desublimed at separate chilled take-off stations. Tails material is desublimed into 48Y cylinders. Product material is desublimed into either 48Y or smaller 30B cylinders.

With the exception of liquid sampling operations, the entire enrichment process operates at subatmospheric pressure. This safety feature helps ensure that releases of UF_6 or HF are minimized because leakage would typically be inward to the system. During sampling operations, UF_6 is liquefied within an autoclave which provides the heating required to homogenize the material for sampling. The autoclave is a rated pressure vessel which serves as secondary containment for the UF_6 product cylinders while the UF_6 is in a liquid state.

There are numerous subsystems associated with each of the major enrichment process systems as well as other facility support and utility systems. These include systems supporting venting, cooling, electrical power, air and water supply, instrumentation and control and handling functions among others.

6.2.3 Process System Descriptions

Detailed system descriptions and design information for enrichment process and process support systems are provided in the NEF Integrated Safety Analysis Summary. These descriptions include information on process technology including materials of construction, process parameters (e.g., flow, temperature, pressure, etc.), key instrumentation and control including alarms/interlocks, and items relied on for safety (IROFS).

6.2.4 Utility and Support System Descriptions

The UF_6 Enrichment Systems also interface with a number of supporting utility systems. Detailed system descriptions and design information for these utility and support systems are provided in the NEF Integrated Safety Analysis Summary. These descriptions include information on process technology including materials of construction; process parameters (e.g., flow, temperature, pressure, etc.), key instrumentation and control including alarms/interlocks, and (IROFS).

6.2.5 Safety Features

There are a number of safety features in place to help prevent, detect, and mitigate potential releases of UF_6. Some of these features are classified as (IROFS) as determined in the Integrated Safety Analysis (ISA). A listing of IROFS associated with process, utility and supporting systems as well as those applicable to the facility and its operations (e.g., administrative controls) is presented in the NEF Integrated Safety Analysis Summary.

In addition to IROFS, there are other process system features that are intended to protect systems from damage that would result in an economic loss. Many of these features have a secondary benefit of enhancing safety by detecting, alarming, and/or interlocking process equipment – either prior to or subsequent to failures that result in a release of material.

6.3 CHEMICAL HAZARDS ANALYSIS

6.3.1 Integrated Safety Analysis

LES has prepared an Integrated Safety Analysis (ISA) as required under 10 CFR 70.62 (CFR, 2003c). The ISA:

- Provides a list of the accident sequences which have the potential to result in radiological and non-radiological releases of chemicals of concern
- Provides reasonable estimates for the likelihood and consequences of each accident identified
- Applies acceptable methods to estimate potential impacts of accidental releases.

The ISA also:

- Identifies adequate engineering and/or administrative controls (IROFS) for each accident sequence of significance
- Satisfies principles of the baseline design criteria and performance requirements in 10 CFR 70.61 (CFR, 2003b) by applying defense-in-depth to high risk chemical release scenarios
- Assures adequate levels of these controls are provided so those items relied on for safety (IROFS) will satisfactorily perform their safety functions.

The ISA demonstrates that the facility and its operations have adequate engineering and/or administrative controls in place to prevent or mitigate high and intermediate consequences from the accident sequences identified and analyzed.

6.3.2 Consequence Analysis Methodology

This section describes the methodology used to determine chemical exposure/dose and radiochemical exposure/dose criteria used to evaluate potential impact to the workers and the public in the event of material release. This section limits itself to the potential effects associated with accidental release conditions. Potential impacts from chronic (e.g., long-term) discharges from the facility are detailed in the Environmental Report.

6.3.2.1 Defining Consequence Severity Categories

The accident sequences identified by the ISA need to be categorized into one of three consequence categories (high, intermediate, or low) based on their forecast radiological, chemical, and/or environmental impacts. Section 6.1.1, Chemical Screening and Classification, presented the radiological and chemical consequence severity limits defined by 10 CFR 70.61 (CFR, 2003b) for the high and intermediate consequence categories.

To quantify criteria of 10 CFR 70.61 (CFR, 2003b) for chemical exposure, standards for each applicable hazardous chemical must be applied to determine exposure that could: (a) endanger the life of a worker; (b) lead to irreversible or other serious long-lasting health effects to an individual; and (c) cause mild transient health effects to an individual. Per NUREG-1520 (NRC 2002), acceptable exposure standards include the Emergency Response Planning Guidelines (ERPG) established by the American Industrial Hygiene Association and the Acute Exposure Guideline Levels (AEGL) established by the National Advisory Committee for Acute Guideline Levels for Hazardous Substances. The definitions of various ERPG and AEGL levels are contained in Table 6.3-1, ERPG and AEGL Level Definitions.

The consequence severity limits of 10 CFR 70.61 (CFR, 2003b) have been summarized and presented in Table 6.3-2, Licensed Material Chemical Consequence Categories. The severity limits defined in this table are developed against set criteria.

The toxicity of UF_6 is due to its two hydrolysis products, HF and UO_2F_2. The toxicological effects of UF_6 as well as these byproducts were previously described in Section 6.1.2. AEGL and NUREG-1391 (NRC, 1991) values for HF and UF_6 were utilized for evaluation of chemotoxic exposure. Additionally, since the byproduct uranyl fluoride is a soluble uranium compound, the AEGL values were derived for evaluating soluble uranium (U) exposure in terms of both chemical toxicity and radiological dose. In general, the chemotoxicity of uranium inhalation/ingestions is of more significance than radiation dose resulting from internal U exposure. The ERPG and AEGL values for HF are presented in Table 6.3-3, ERPG and AEGL values for Hydrogen Fluoride. The ERPG and AEGL values for UF_6 (as soluble U) are presented in Table 6.3-4, ERPG and AEGL values for Uranium Hexafluoride (as soluble U). The values from NUREG-1391 (NRC, 1991) for soluble uranium are presented in Table 6.3-6, Health Effects from Intake of Soluble Uranium.

Table 6.3-5, Definition of Consequence Severity Categories, presents values for HF and UF_6 (as soluble U) from the AEGL and NUREG-1391 (NRC, 1991).

6.3.2.1.1 Worker Exposure Assumptions

Any release from UF_6 systems/cylinders at the facility would predominantly consist of HF with some potential entrainment of uranic particulate. An HF release would cause a visible cloud and a pungent odor. The odor threshold for HF is less than 1 ppm and the irritating effects of HF are intolerable at concentrations well below those that could cause permanent injury or which produce escape-impairing symptoms. Employees are trained in proper actions to take in response to a release and it can be confidently predicted that workers will take immediate self-protective action to escape a release area upon detecting any significant HF odor.

For the purposes of evaluating worker exposure in cases where a local worker would be expected to be in the immediate proximity of a release (e.g., connect/disconnect, maintenance, etc.), the 10-minute AEGL values have been used for HF and NUREG-1391 (NRC, 1991) values have been used for U. In these cases, it has been presumed that the operator will fail to recognize the in-rush of air into the vacuum system and will not begin to back away from the source of the leak until HF is present. Sufficient time is available for the worker to reliably detect and evacuate the area of concern.

For the purposes of evaluating worker exposures for workers who may be present elsewhere in the room of release, the values in Table 6.3-5, Definition of Consequence Severity Categories, which are the 10-minute AEGL values, have been used. Once a release is detected the worker is assumed to evacuate the area of concern. Sufficient time is available for the worker to reliably detect and evacuate the area of concern.

Another assumption made in conducting consequence severity analysis is that for releases precipitated by a fire event, only public exposure was considered in determining consequence severity; worker exposures were not considered. The worker is assumed to evacuate the area of concern once the fire is detected by the worker. Fires of sufficient magnitude to generate chemical/radiological release must either have caused failure of a mechanical system/component or involve substantive combustibles containing uranic content. In either case, the space would be untenable for unprotected workers. Sufficient time is available for the worker to reliably detect and evacuate the area of concern prior to any release. Fire brigade/fire department members responding to emergencies are required by emergency response procedure (and regulation) to have suitable respiratory and personal protective equipment.

6.3.2.1.2 Public Exposure Assumptions

Potential exposures to members of the public were also evaluated assuming conservative assumptions for both exposure concentrations and durations. Exposure was evaluated for consequence severity against chemotoxic, radiotoxic, and radiological dose.

Public exposures were estimated to last for a duration of 30 minutes. This is consistent with self-protective criteria for UF_6/HF plumes listed in NUREG-1140 (NRC, 1988).

6.3.2.2 Chemical Release Scenarios

The evaluation level chemical release scenarios based on the criteria applied in the Integrated Safety Analysis are presented in the NEF Integrated Safety Analysis Summary. Information on the criteria for the development of these scenarios is also provided in the NEF Integrated Safety Analysis Summary.

6.3.2.3 Source Term

The methodologies used to determine source term are those prescribed in NUREG/CR-6410 (NRC, 1998) and supporting documents.

6.3.2.3.1 Dispersion Methodology

In estimating the dispersion of chemical releases from the facility, conservative dispersion methodologies were utilized. Site boundary atmospheric dispersion factors were generated using a computer code based on Regulatory Guide 1.145 (NRC, 1982) methodology. The code was executed using five years (1987-1991) of meteorological data collected at Midland/Odessa, Texas, which is the closest first order National Weather Service Station to the site. This station was judged to be representative of the NEF site because the Midland Odessa National Weather Service Station site and the NEF site have similar climates and topography.

The specific modeling methods utilized follow consistent and conservative methods for source term determination, release fraction, dispersion factors, and meteorological conditions as prescribed in NRC Regulatory Guide 1.145 (NRC, 1982).

For releases inside of buildings, conservative leak path fractions were assumed as recommended by NUREG/CR-6410 (NRC, 1998) and ventilation on and off cases were evaluated for consideration of volumetric dilution and mixing efficiency prior to release to atmosphere.

6.3.2.4 Chemical Hazard Evaluation

This section is focused on presenting potential deleterious effects that might occur as a result of chemical release from the facility. As required by 10 CFR 70 (CFR, 2003a), the likelihood of these accidental releases fall into either unlikely or highly unlikely categories.

6.3.2.4.1 Potential Effects to Workers/Public

The toxicological properties of potential chemicals of concern were detailed in Section 6.2, Chemical Process Information. The evaluation level accident scenarios identified in the Integrated Safety Analysis and the associated potential consequence severities to facility workers or members of the public are presented in the NEF Integrated Safety Analysis Summary.

All postulated incidents have been determined to present low consequences to the workers/public, or where determined to have the potential for intermediate or high consequences, are protected with IROFS to values less than the likelihood thresholds required by 10 CFR 70.61 (CFR, 2003b).

6.3.2.4.2 Potential Effects to Facility

All postulated incidents have been determined to present inherently low consequences to the facility. No individual incident scenarios were identified that propagate additional consequence to the facility process systems or process equipment. The impact of external events on the facility, and their ability to impact process systems or equipment of concern is discussed in the NEF Integrated Safety Analysis Summary.

6.4 CHEMICAL SAFETY ASSURANCE

The facility will be designed, constructed and operated such that injurious chemical release events are prevented. Chemical process safety at the facility is assured by designing the structures, systems and components with safety margins such that safe conditions are maintained under normal and abnormal process conditions and during any credible accident or external event.

6.4.1 Management Structure and Concepts

The criteria used for chemical process safety encompasses principles stated in NUREG-1601, Chemical Process Safety at Fuel Cycle Facilities (NRC, 1997). It is also supported by concepts advocated in 29 CFR 1910.119, Process Safety Management of Highly Hazardous Chemicals (CFR, 2003f), and 40 CFR, 68, Accidental Release Prevention Requirements (CFR, 2003g), although it is noted here that there are no chemicals at this facility which exceed threshold planning quantities of either standard.

The intent of chemical safety management principles is to identify, evaluate, and control the risk of chemical release through engineered, administrative, and related safeguards.

The chemical safety philosophy for the facility is to apply sufficient control to identify, evaluate, and control the risk of accidental chemical releases associated with licensed material production to acceptable levels in accordance with 10 CFR 70.61(b) and (c) (CFR, 2003b).

The identification and evaluation of chemical release risk has been developed through the conduct of an ISA. The development of these scenarios, and the dispersion analysis and chemical/radiological dose assessment associated with each accident sequence was performed and was conducted in accordance with NUREG/CR-6410, Nuclear Fuel Cycle Facility Accident Analysis Handbook (NRC, 1998) as was described previously in Section 6.3, Chemical Hazards Analysis.

The control of chemical release risk is ensured through numerous features that are described in the following sections.

6.4.2 System Design

The design of chemical process systems includes numerous controls for maintaining safe conditions during process operations. This is accomplished through several means including managing the arrangement and size of material containers and processes, selection and use of materials compatible with process chemicals, providing inherently safer operating conditions (e.g., vacuum handling), providing process interlocks, controls, and alarming within the chemical processes. All of these plant and equipment features help assure prevention of chemical release. Process piping and components, (e.g., centrifuges, traps, vents, etc.) are maintained safe by limits placed on their operating parameters.

With respect to chemical process safety design features recommended in NUREG-1601 (NRC, 1997), this section briefly details the features provided for the UF_6 system which is the only chemical of concern (Class 1) process system.

6.4.2.1 Physical Barriers

Double-Walled Piping and Tanks - The UF_6 system piping operates at subatmospheric pressure throughout the plant except for the liquid sampling operation which is conducted within a secondary containment autoclave. As such, UF_6 system piping is not double-walled. Criticality design has been addressed for this vessel.

Liquid Confinement Dikes – Dikes are provided in areas where uranic material is present in solution in tankage. Criticality design constraints were applied to these containment areas. Confinement dikes are also present for chemical spillage control in TSB areas.

Glove Boxes – Glove boxes are utilized for a small number of decontamination operations (e.g., sample bottles, flex hoses). They are not needed for other operations as the levels of specific activity are low. To confine potential HF/uranic material effluent, flexible exhaust hoses connected to the GEVS are provided for locations where UF_6 systems will be opened (e.g., hose connect/disconnect, maintenance, etc.) to capture any fumes remaining after purging operations. GEVS flexible exhaust hoses and fume hoods are present in the TSB where uranic material containers are opened during laboratory and waste handling operations.

Splash Shields – There are no areas where bulk liquid hazardous chemicals will be handled. Lab operations with hazardous chemicals will be conducted in hoods and/or with appropriate personnel protective equipment for these small-scale operations.

Fire Walls – Fire walls are provided to separate UF_6 and uranic material handling areas from other areas of the facility.

Protective Cages – Protective barriers are provided to protect UF_6 system susceptible components (e.g., piping, small equipment) in areas where there is major traffic.

Backflow Preventers and Siphon Breaks – Liquid systems with high uranic content (i.e., not trace waste streams) are provided with means to prevent backflow or siphon. For the UF_6 gaseous piping, design features are provided to prevent UF_6 migration into the few systems which are required to be interconnected to UF_6.

Overflow vessel – UF_6 is not handled in liquid form in any continuous process and any batch handling is performed in small lab quantities or in a secondary containment autoclave. For those systems where uranic material is in solution, overflow protection features are provided.

Chemical Traps and Filters - Chemical traps and filters are provided on vent and ventilation systems which capture UF_6 to remove HF and uranic contaminants prior to any discharge to atmosphere.

6.4.2.2 Mitigative Features

Driving Force Controls – Driving force controls are provided to isolate heating/cooling equipment at UF_6 take-off stations and cold traps as well as other uranic material containing systems. Other driving force controls include relief valves and cut-offs on the nitrogen system to protect the UF_6 system from overpressure.

Solenoid and Control Valves – These types of valves are provided to stop and/or regulate the flow of UF_6 in the event of abnormal operating conditions.

Spray Systems – Spray systems are not provided for UF_6 systems or system areas due to criticality control requirements.

Alarm Systems – Alarm systems are provided which will alarm in the Control Room for abnormal process parameter (e.g., flow, temperature, pressure, level, etc.) conditions in the UF_6 system and some supporting systems. Leak detection is also provided to detect the release of UF_6/HF in the facility GEVS systems and other ventilation systems. Alarm measures are in place to notify facility employees of the need to evacuate process areas and/or the facility in the event of a serious chemical release.

6.4.2.3 Baseline Design Criteria and Defense-In-Depth

The ISA demonstrates that the design and construction complies with the baseline design criteria (BDC) of 10 CFR 70.64(a) (CFR, 2003d), and the defense-in-depth requirements of 10 CFR 70.64(b) (CFR, 2003d). The design provides for adequate protection against chemical risks produced from licensed material, facility conditions which affect the safety of licensed material, and hazardous chemicals produced from licensed material. The NEF is not proposing any facility-specific or process-specific relaxation or additions to applicable BDC features.

6.4.3 Configuration Management

Configuration management includes those controls which ensure that the facility design basis is thoroughly documented and maintained, and that changes to the design basis are controlled. This includes the following:

A. That management commitment and staffing is appropriate to ensure configuration management is maintained

B. That proper quality assurance (QA) is in place for design control, document control, and records management

C. That all structures, systems, and components, including IROFS, are under appropriate configuration management.

A more detailed description of the configuration management system can be found in Section 11.1, Configuration Management (CM).

6.4.4 Maintenance

The NEF helps maintain chemical process safety through the implementation of administrative controls that ensure that process system integrity is maintained and that IROFS and other engineered controls are available and operate reliably. These controls include planned and scheduled maintenance of equipment and controls so that design features will function when required. Appropriate plant management is responsible for ensuring the operational readiness of IROFS under this control. For this reason, the maintenance function is closely coupled to operations. The maintenance function plans, schedules, tracks, and maintains records for maintenance activities.

Maintenance activities generally fall into the following categories:

A. Surveillance/monitoring

B. Corrective maintenance

C. Preventive maintenance

D. Functional testing.

A more detailed description of the maintenance program and maintenance management system can be found in Section 11.2, Maintenance.

6.4.5 Training

Training in chemical process safety is provided to individuals who handle licensed materials and other chemicals at the facility. The training program is developed and implemented with input from the chemical safety staff, training staff, and management. The program includes the following:

A. Analysis of jobs and tasks to determine what a worker must know to perform tasks efficiently

B. Design and development of learning objectives based upon the analysis of jobs and tasks that reflect the knowledge, skills, and abilities needed by the worker

C. Design and development of qualification requirements for positions where a level of technical capability must be achieved and demonstrated for safe and reliable performance of the job function

D. Development and implementation of standard and temporary operating procedures

E. Development and implementation of proper inspection, test, and maintenance programs and procedures

F. Development of chemical safety awareness throughout the facility so that all individuals know what their roles and responsibilities are in coordinating chemical release mitigation activities - in support of the Emergency Plan - in the event of a severe chemical release

G. Coordination of chemical process safety training curriculum with that of other areas including, radiological safety, criticality safety, facility operations, emergency response, and related areas.

A more detailed description of the training program can be found in Section 11.3, Training and Qualifications.

6.4.6 Procedures

A key element of chemical process safety is the development and implementation of procedures that help ensure reliable and safe operation of chemical process systems.

Generally, four types of plant procedures are used to control activities: operating procedures, administrative procedures, maintenance procedures, and emergency procedures.

Operating procedures, developed for workstation and Control Room operators, are used to directly control process operations. Operating procedures include:

- Directions for normal operations, including startup and some testing, operation, and shutdown, as well as off-normal conditions of operation, including alarm response
- Required actions to ensure radiological and nuclear criticality safety, chemical safety, fire protection, emergency planning, and environmental protection
- Operating limits, controls and specific direction regarding administrative controls to ensure operational safety
- Safety checkpoints such as hold points for radiological or criticality safety checks, QA verifications, or operator independent verification.

Administrative procedures are used to perform activities that support the process operations, including, but not limited to, management measures such as the following:

- Configuration management
- Nuclear criticality, radiation, chemical, and fire safety
- Quality assurance
- Design control
- Plant personnel training and qualification
- Audits and assessments
- Incident investigations
- Record keeping and document control
- Reporting.

Administrative procedures are also used for:

- Implementing the Fundamental Nuclear Material Control (FNMC) Plan
- Implementing the Emergency Plan
- Implementing the Physical Security Plan
- Implementing the Standard Practice Procedures Plan for the Protection of Classified Matter.

Maintenance procedures address:

- Preventive and corrective maintenance of IROFS
- Surveillance (includes calibration, inspection, and other surveillance testing)
- Functional testing of IROFS
- Requirements for pre-maintenance activity involving reviews of the work to be performed and reviews of procedures.

Emergency procedures address the preplanned actions of operators and other plant personnel in the event of an emergency.

A more detailed description of the procedural development and management program can be found in Section 11.4, Procedures Development and Implementation.

6.4.7 Chemical Safety Audits

Audits are conducted to determine that plant operations are performed in compliance with regulatory requirements, license conditions, and written procedures. As a minimum, they assess activities related to radiation protection, criticality safety control, hazardous chemical safety, fire protection, and environmental protection.

Audits are performed in accordance with a written plan, which identifies and schedules audits to be performed. Audit team members shall not have direct responsibility for the function and area being audited. Team members have technical expertise or experience in the area being audited and are indoctrinated in audit techniques. Audits are conducted on an annual basis on select functions and areas as defined above. The chemical process safety functions and areas will be audited at least triennially.

Qualified staff personnel that are not directly responsible for production activities are utilized to perform routine surveillances/assessments. Deficiencies noted during the inspection requiring corrective action are forwarded to the manager of the applicable area or function for action. Future surveillances/assessments include a review to evaluate if corrective actions have been effective.

A more detailed description of the audit program can be found in Section 11.5, Audits and Assessments.

6.4.8 Emergency Planning

The NEF has a facility emergency plan and program which includes response to mitigate the potential impact of any process chemical release including requirements for notification and reporting of accidental chemical releases.

The LES fire brigade/emergency response team is outfitted, equipped, and trained to provide hazardous material response and mitigation commensurate with the requirements of 29 CFR 1910.120, Hazardous waste operations and emergency response (CFR, 2004). This includes a technician level qualified entry and backup team with supporting emergency medical function, incident command, and a safety officer. The safety officer has the additional responsibility to monitor response activities to ensure that criticality safety is maintained.

The City of Hobbs, NM Fire Department is the nearest offsite response agency who can supplement LES with additional Hazardous Waste Operations and Emergency Response (HAZWOPER) response teams. As a result of a baseline needs assessment conducted on offsite response, LES has committed to assist the local offsite fire agency, Eunice Fire and Rescue, in obtaining the equipment and training to also provide a HAZWOPER compliant response team.

Additional information on emergency response can be found in SAR Section 7.5.2, Fire Emergency Response, and in the NEF Emergency Plan.

6.4.9 Incident Investigation and Corrective Actions

A facility wide incident investigation process exists that includes chemical process related incidents. This process is available for use by any person at the facility for reporting abnormal events and potentially unsafe conditions or activities. Abnormal events that potentially threaten or lessen the effectiveness of health, safety or environmental protection will be identified and reported to and investigated by the Health, Safety, and Environment (HS&E) Manager. Each event will be considered in terms of its requirements for reporting in accordance with regulations and will be evaluated to determine the level of investigation required. These evaluations and investigations will be conducted in accordance with approved procedures. The depth of the investigation will depend upon the severity of the classified incident in terms of the levels of uranium/chemical released and/or the degree of potential for exposure of workers, the public or the environment.

A more detailed description of the incident investigation program can be found in Section 11.6, Incident Investigations and Corrective Action Process.

6.5 REFERENCES

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TABLES

Table 6.1-1 Chemicals – Hazardous Properties

Form	Chemical	Class	Chemical Formula	Corrosive	Flammable	Combustible	Oxidizer	Reactive	Toxic	Radioactive	Health Hazard	Irritant	Remarks
Liquid	uranium hexafluoride	1	UF_6	✓				✓	✓	✓			
	uranium compounds (residual)		UO_2F_2						✓	✓			Byproduct – no NEF class
	silicone oil	2				✓							
	ethanol	3	C_2H_5OH		✓								
	methylene chloride	3	CH_2Cl_2								✓		
	oil	3				✓							
	cutting oil	3				✓							
	paint	3				✓							
	degreaser solvent, SS25	3				✓							
	penetrating oil	3				✓							
	PFPE (Tyreno) oil	2											Note 3
	organic chemicals	3			✓								
	nitric acid (65%)	3	HNO_3	✓									
	hydrogen peroxide	3	H_2O_2				✓						
	acetone	3	C_3H_6O		✓								
	toluene	3	C_7H_8		✓								
	petroleum ether	3			✓								
	sulfuric acid	3	H_2SO_4	✓									
	phosphoric acid	3	H_3PO_4	✓									
	sodium hydroxide (0.1N)	3	NaOH	✓									
	diesel fuel (outdoor)	3				✓							
	laboratory effluent (aqueous)	2											Note 1
	citric acid waste	2											Note 1
	precipitation sludge	3											Note 1
	evaporator/dryer sludge	2											Note 1
	hand wash / shower water	3											Note 1
	miscellaneous samples	3											Note 1 & 2
	R23 trifluoromethane	2	CHF_3										Note 3
	R404A fluoroethane blend	2	C_2HF_5 / $C_2H_3F_3$ / $C_2H_2F_4$										Note 3

Table 6.1-1 Chemicals – Hazardous Properties
Page 2 of 3

Form	Chemical	Class	Chemical Formula	Corrosive	Flammable	Combustible	Oxidizer	Reactive	Toxic	Radioactive	Health Hazard	Irritant	Remarks
	R507 penta/tri fluoroethane	2	C_2HF_5 / $C_2H_3F_3$										Note 3
	detergent	3											Note 3
	laundry effluent water	3											Note 1
	PFPE (Fomblin) oil	2											Note 3
	floor wash water	3											Note 1
	citric acid, 5-10%	2											Note 3
	degreaser water	3											Note 1
	degreaser sludge	3											Note 1
	standard solutions	3											Note 2
	urine	3											Note 3
	nitrogen	2	N_2										Note 3
	miscellaneous chemicals (utilities)	3											Note 2
	potassium or sodium hydroxide	3	KOH/NaOH	✓									
	hydrocarbon sludge	3				✓							
Gas	uranium hexafluoride	1	UF_6	✓				✓	✓	✓			
	uranium compounds		UO_2F_2						✓	✓			Byproduct – no NEF class
	hydrogen fluoride		HF	✓					✓				Byproduct – no NEF class
	oxygen gas	3	O_2				✓						
	acetylene gas	3	C_2H_2		✓								
	propane gas	3	C_3H_8		✓								
	primus gas	3	C_4H_{10} / C_3H_7		✓								
	hydrogen	3	H_2		✓								
	R23 trifluoromethane	2	CHF_3										Note 3
	R404A fluoroethane blend	2	C_2HF_5 / $C_2H_3F_3$ / $C_2H_2F_4$										Note 3
	R507 penta/tri fluoroethane	2	C_2HF_5 / $C_2H_3F_3$										Note 3
	helium	2	He										Note 3
	argon	3	Ar										Note 3
	nitrogen	2	N_2										Note 3

Table 6.1-1 Chemicals – Hazardous Properties

Form	Chemical	Class	Chemical Formula	Corrosive	Flammable	Combustible	Oxidizer	Reactive	Toxic	Radioactive	Health Hazard	Irritant	Remarks
Solid	uranium hexafluoride	1	UF_6	✓				✓	✓	✓			
	sodium fluoride	2	NaF						✓				Note 1
	papers, wipes, gloves, etc.	3				✓							Note 1
	contaminated disposable clothing	3				✓							Note 1
	laundry	3				✓							Note 1
	uranium compounds	3	UO_2F_2						✓	✓			
	combustible solid waste	3				✓							Note 1
	citric acid, crystalline	3	$C_6H_8O_4$									✓	
	activated carbon	2	C										Note 1
	aluminum oxide	2	Al_2O_3										Note 1
	carbon fibers	3											Note 1
	metals (aluminum)	3											Note 3
	sand blasting sand	3											Note 3
	shot blaster media	3											Note 3
	ion exchange resin	3											Note 1
	filters, radioactive	3											Note 1
	filters, industrial	3											Note 3
	carbon/potassium carbonate	3											Note 1
	soils and grass	3											Note 3
	diatomaceous earth (celite)	3									✓	✓	
	sodium carbonate	2	Na_2CO_3						✓			✓	
	scrap metals	3								✓			
	non-metallic waste (plastic)	3								✓			

NOTES

1. Many waste streams including gaseous effluent, liquid waste, and solid waste will contain some level of residual uranium compounds, not within toxic concentrations. The radiation hazard is listed separately from these chemicals as residual uranium compounds.

2. Each component in the miscellaneous samples and standard solutions, in the chemical laboratory, is not specified.

3. These chemicals do not fall under any of the listed hazard categories.

Table 6.1-2 Chemicals – Separations Building

CHEMICAL/PRODUCT			INVENTORY BY LOCATION								REMARKS
NAME	FORMULA	PHYSICAL STATE	UBC STORAGE PAD (outdoors) – see Note 4	CYLINDER RECEIPT & DISPATCH BUILDING (CRDB)	UF_6 HANDLING AREA	CASCADE HALLS	FIRST FLOOR PROCESS SERVICES AREA	SECOND FLOOR PROCESS SERVICES AREA	THIRD FLOOR PROCESS SERVICES AREA	BLENDING AND LIQUID SAMPLING AREA	
							No chemicals		No chemicals		
uranium hexafluoride	UF_6	*solid*	1.97E8 kg (4.34E6 lb)	9.43E6 kg (2.08E7 lb)	4.00E5 kg/module (8.82E5 lb/ module)					1.34E5 kg (2.95E5 lb)	Notes 1, 2, 3, & 4
uranium hexafluoride	UF_6	liquid								1.15E4 kg (2.54E4 lb)	Note 2
uranium hexafluoride	UF_6	gas				256 kg/module (565 lb/module)		13.8 kg/module (30.4 lb/ module)		3 kg/module (6.6 lb/module)	Note 5
hydrogen fluoride	HF	gas			piping (trace)						
silicone oil		liquid			560 L / module (148 gal/module)					70 L (18.5 (gal)	
sodium fluoride	NaF	solid						4800 kg/module (10,584 lb/ module			
R23 *trifluoromethane*		gas/liquid			13.6 kg/module 30.0 lb/module)					1.7 kg (3.7 lb)	
R404A fluoroethane blend		gas/liquid			120 kg/module (265 lb/module)					15 kg (33.1 lb)	
R507 penta/tri fluoroethane		gas/liquid			510 kg/module 1125 lb/module)					60 kg (132 lb)	
activated carbon	C	granules			624 kg (1376 lb)					13 kg (28.7 lb)	
aluminum oxide	Al_2O_3	granules			828 kg (1826 lb)					23 kg (50.7 lb)	

NOTES:

1. The CRDB can house up to 708 feed cylinders 122 cm(48 in) diameter, 125 product cylinders 76 cm (30 in) diameter, and 125 semi-finished product cylinders 76 cm (30 in) diameter
2. The Blending and Liquid Sampling Area can have up to 8 (48Y) cylinders in storage/transition, 2 (48Y) cylinders in donor stations, 4 (30B) cylinders in receiver stations. Up to 5 (30B) cylinders can be present in liquid sampling autoclaves and will be in various physical states depending on sampling in progress.
3. UF_6 Handling Area inventory is maximum estimated operational inventory.
4. The UBC Storage Pad is located outside of and detached from the Separations Building.
5. Normal estimated operational inventory in piping. Gas flows in piping routed from the UF6 Handling Area to the Cascade Halls and back. The Process Services Area contains the main manifolds and valve stations.

Table 6.1-3 Chemicals -- Centrifuge Assembly Building

CHEMICAL/PRODUCT			INVENTORY BY LOCATION			REMARKS
NAME	FORMULA	PHYSICAL STATE	CENTRIFUGE ASSEMBLY AREA	CENTRIFUGE TEST FACILITY	CENTRIFUGE POST MORTEM FACILITY	
ethanol	C_2H_6O	liquid	40 L (10.6 gal)			Note 1
methylene chloride	CH_2Cl_2	liquid	40 L (10.6 gal)			Note 1
uranium hexafluoride	UF_6	gas/solid		50kg (110 lb)	Residual	Notes 2 & 3
helium	He	gas	440 m^3 (15536 ft^3)			Gas volume is at Std. Conditions.
argon	Ar	gas	190 m^3 (6709 ft^3)			Gas volume is at Std. Conditions.
activated carbon	C	granules		10 kg (22.1 lb)		
aluminum oxide	Al_2O_3	granules		20 kg (44.1 lb)		

NOTES:
1. In the Centrifuge Assembly Area, ethanol and methylene chloride are used as cleaning agents. Total quantity of both solvents used in one year is 80 L (21.2 gal).
2. Centrifuges in the Centrifuge Post Mortem Facility are considered contaminated based on previous operation with UF_6. Once in the Centrifuge Post Mortem Facility they will not contain significant amounts of UF_6.
3. In the Centrifuge Test Facility 50 kg (110 lb) of UF_6 is contained in a feed vessel, test centrifuges, and a take-off vessel. Physical state will vary depending on testing in progress.

Table 6.1-4 Chemicals – Technical Services Building

CHEMICAL/PRODUCT			INVENTORY BY LOCATION												REMARKS
NAME	FORMULA	PHYSICAL STATE	LAUNDRY SYSTEM	VENTILATED ROOM	DECONTAMINATION WORKSHOP	ME & I WORKSHOP	VACUUM PUMP REBUILD WORKSHOP	LIQUID EFFLUENT COLLECTION AND TREATMENT SYSTEM	SOLID WASTE COLLECTION SYSTEM	GASEOUS EFFLUENT VENT SYSTEM (TSB)	CYLINDER PREPARATION ROOM	CHEMICAL LABORATORY	ENVIRONMENTAL MONITORING LABORATORY	MASS SPECTROMETRY LABORATORY	
uranium hexafluoride	UF_6	solid		2300-12500 kg (5071-27563 lb)	residual							250 kg (551 lb)		0.5 kg (1.1 lb)	
sodium fluoride	NaF	powder					100 kg (221 lb)								
oxygen gas	O_2	gas				11 m^3 (388 ft^3)									
acetylene gas	C_2H_2	gas				6 m^3 (212 ft^3)									
propane gas	C_3H_8	gas				0.68 kg (1.50 lb)									
cutting oil		liquid				2.4 L (0.6 gal)	0.08 kg (0.18 lb)								
paint		liquid				2.4 L (0.6 gal)	9.6 L (2.5 gal)								
primus gas		gas					0.5 kg (1.1 lb)								
degreaser solvent, SS25		liquid					2.4 L (0.6 gal)								
penetrating oil		liquid					0.44 L (0.12 gal)								

CHEMICAL/PRODUCT			INVENTORY BY LOCATION												REMARKS
NAME	FORMULA	PHYSICAL STATE	LAUNDRY SYSTEM	VENTILATED ROOM	DECONTAMINATION WORKSHOP	ME & I WORKSHOP	VACUUM PUMP REBUILD WORKSHOP	LIQUID EFFLUENT COLLECTION AND TREATMENT SYSTEM	SOLID WASTE COLLECTION SYSTEM	GASEOUS EFFLUENT VENT SYSTEM (TSB)	CYLINDER PREPARATION ROOM	CHEMICAL LABORATORY	ENVIRONMENTAL MONITORING LABORATORY	MASS SPECTROMETRY LABORATORY	
PFPE (Tyreno) oil		liquid					120 L (31.7 gal)								
organic chemicals		liquid							50 L (13.2 gal)						
potassium or sodium hydroxide	KOH/NaOH	liquid						210 L (55.4 gal)							
nitric acid (65%)	HNO_3	liquid										26 L (6.9 gal)			
ethanol (100%)	C_2H_6O	liquid										5 L (1.3 gal)			
peroxide	H_2O_2	liquid										4 L (1.1 gal)			
acetone	C_3H_6O	liquid										27 L 7.1 gal)			
toluene	C_7H_8	liquid										2 L (0.5 gal)			
petroleum ether		liquid										10 L (2.6 gal)			
sulfuric acid	H_2SO_4	liquid										10 L 2.6 gal)			
phosphoric acid	H_3PO_4	liquid										44 L (11.6 gal)			
sodium hydroxide (0.1N)	NaOH	liquid										5 L (1.3 gal)			
methylene chloride	CH_2Cl_2	liquid					210 L 55.4 gal)		420 L (111 gal)						
hydrogen	H_2	gas											std. cylinder		

Table 6.1-4 Chemicals – Technical Services Building
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CHEMICAL/PRODUCT			INVENTORY BY LOCATION													REMARKS
NAME	FORMULA	PHYSICAL STATE	LAUNDRY SYSTEM	VENTILATED ROOM	DECONTAMINATION WORKSHOP	ME & I WORKSHOP	VACUUM PUMP REBUILD WORKSHOP	LIQUID EFFLUENT COLLECTION AND TREATMENT SYSTEM	SOLID WASTE COLLECTION SYSTEM	GASEOUS EFFLUENT VENT SYSTEM (TSB)	CYLINDER PREPARATION ROOM	CHEMICAL LABORATORY	ENVIRONMENTAL MONITORING LABORATORY	MASS SPECTROMETRY LABORATORY	
PFPE (Fomblin) oil		liquid			10 L (2.6 gal)		10 L (2.6 gal)								
activated carbon	C	granules		10 kg & 210 L (22.1 lb & 55.4 gal)			10 kg (22.1 lb)		50 kg (110 lb)		13 kg (28.7 lb)				
aluminum oxide	Al_2O_3	granules		40 kg & 210 L (88.2 lb & 55.4 gal)			20 kg (44.1 lb)		360 kg (794 lb)		23 kg (50.7 lb)				
citric acid, 5-10%		solution			800 L (211 gal)										
citric acid, waste		solution						1325 L (350 gal)							
gaseous nitrogen	N_2	gas		piping		10 m^3 (353 ft^3)						piping	piping		
ion exchange resin		solid						0.8 m^3 (28.2 ft^3)	0.8 m^3 (28.2 ft^3)						
carbon/potassium carbonate		granules								filter					
argon	Ar	gas												190 L (50.2 gal)	
liquid nitrogen	N_2	liquid										2 L (0.5 gal)			
diatomaceous earth		powder			10kg (22.1 lb)										
sodium carbonate	Na_2CO_3	granules			10kg (22.1 lb)										

Table 6.1-5 Chemicals – Central Utilities Building

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CHEMICAL/PRODUCT			INVENTORY BY LOCATION			REMARKS
NAME	FORMULA	PHYSICAL STATE	NITROGEN SYSTEM	ADDITIONAL UTILITIES SYSTEMS	ELECTRICAL SYSTEM	
Diesel fuel (outdoors)		liquid			37,854 L (10,000 gal)	2 tanks at 18,927 L (5,000 gal) each
cryogenic nitrogen (outdoors)	N_2	liquid	37,856 L (10,000 gal)			4 tanks at 9,464 L (2,500 gal) each

Table 6.1-6 Physical Properties of UF_6

Property	Value
Sublimation Point at 1.01 bar abs (14.7 psia)	56.6°C (133.8°F)
Triple Point	1.52 bar abs (22 psia) 64.1°C (147.3°F)
Density Solid @ 20°C (68°F) Liquid @ 64.1°C (147.3°F) Liquid @ 93°C (200°F) Liquid @ 113°C (235°F) Liquid @ 121°C (250°F)	 5.1 g/cc (317.8 lb/ft^3) 3.6 g/cc (227.7 lb/ft^3) 3.5 g/cc (215.6 lb/ft^3) 3.3 g/cc (207.1 lb/ft^3) 3.3 g/cc (203.3 lb/ft^3)
Heat of Sublimation @ 64.1°C (147.3°F)	135,373 J/kg (58.2 BTU/lb)
Heat of Fusion @ 64.1°C (147.3°F)	54,661 J/kg (23.5 BTU/lb)
Heat of Vaporization @ 64.1°C (147.3°F)	81,643 J/kg (35.1 BTU/lb)
Specific Heat Solid @ 27°C (81°F) Liquid @ 72°C (162°F)	 477 J/kg/°K (0.114 BTU/lb/°F) 544 J/kg/°K (0.130 BTU/lb/°F)
Critical Pressure	46.10 bar abs (668.8 psia)
Critical Temperature	230.2°C (446.4°F)

Table 6.2-1 Properties of Chemical Adsorbents

Adsorbent (solid)/ Adsorbate (gas)	Heat of Adsorption	Capacity of Adsorption by weight
Activated Carbon/UF_6	293 kJ/kg (126 BTU/lb)	1:1
Activated Carbon/HF	negligible	negligible at low pressure
Aluminum Oxide/UF_6	negligible	0.2:1
Aluminum Oxide/HF	negligible	0.2:1
Activated NaF/UF_6	186 kJ/kg (80 BTU/lb)	1.0-1.5:1
Activated NaF/HF	4,052 kJ/kg (1,742 BTU/lb)	1:0.5

Table 6.2-2 UF_6 Corrosion Rates
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Material	Corrosion Rate @ 20°C (68°F) per year	Corrosion Rate @ 100°C (212°F) per year
Aluminum	6.6E-7 mm (2.6E-5 mils)	8.4E-5 mm (3.3E-3 mils)
Stainless Steel	1.4E-4 mm (5.5E-3 mils)	0.03 mm (1.2 mils)
Copper	1.2E-4 mm (4.7E-3 mils)	3.3E-3 mm (1.3E-1 mils)
Nickel	< 0.05 mm (< 2.0 mils)	< 0.05 mm (< 2.0 mils)

Table 6.2-3 Materials of Construction for UF_6 Systems

Component	Material	Wall Thickness (nominal)	Wall Thickness (minimum)
UF_6 Feed Cylinders (48Y, 48X) and UBCs (48Y)	Carbon Steel ASTM A516	16 mm (0.625 inch)	12.7 mm (0.5 inch)
UF_6 Product Cylinder (30B)	Carbon Steel ASTM A516	12.7 mm (0.5 inch)	8 mm (0.3125 inch)
Sample Bottle (1S)	Nickel/Monel ASTM B162	1.6 mm (0.0625 inch)	1.6 mm (0.0625 inch)
Sample Bottle (2S)	Nickel/Monel ASTM B162	2.8 mm (0.112 inch)	1.6 mm (0.0625 inch)
UF_6 Piping	Aluminum & Stainless Steel	3.7 mm (0.147 inch)	not applicable
UF_6 Valves	Aluminum & Stainless Steel	> 3.7 mm (> 0.147 inch)	not applicable
Cold Trap	Stainless Steel	8 mm (0.315 inch)	not applicable

Table 6.3-1 ERPG and AEGL Level Definitions

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Emergency Response Planning Guideline (ERPG)		Acute Exposure Guideline Level (AEGL)	
General Definition	Values intended to provide estimates of concentration ranges above which one could be responsibly anticipate observing health effects.	**General Definition**	Threshold exposure limits for the protection of the general public, which are applicable to emergency exposure periods ranging from 10 minutes to 8 hours. It is believed that the recommended exposure levels are applicable to general population including infants and children, and other individuals who may be sensitive and susceptible.
ERPG-1	The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to 1 hour without experiencing more than mild, transient adverse health effects or without perceiving a clearly defined objectionable odor.	**AEGL-1** (non-disabling)	The airborne concentration of a substance above which it is predicted that the general population, including susceptible individuals, could experience notable discomfort, irritation or certain asymptomatic, non-sensory effects. However, the effects are not disabling and are transient and reversible upon cessation of exposure.
ERPG-2	The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to 1 hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair an individual's ability to take protective action.	**AEGL-2** (disabling)	The airborne concentration of a substance above which it is predicted that the general population, including susceptible individuals, could experience irreversible or other serious, long-lasting adverse health effects, or an impaired ability to escape.
ERPG-3	The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to 1 hour without experiencing or developing life-threatening health effects.	**AEGL-3** (lethality)	The airborne concentration of a substance above which it is predicted that the general population, including susceptible individuals, could experience life-threatening health effects or death.

Table 6.3-2 Licensed Material Chemical Consequence Categories

	Workers	Offsite Public	Environment
Category 3 High Consequence	Radiation Dose (RD) >1 Sievert (Sv) (100 rem) For the worker (elsewhere in room), except the worker (local), Chemical Dose (CD) > AEGL-3 For worker (local), CD > AEGL-3 for HF CD > * for U	RD > 0.25 Sv (25 rem) 30 mg sol U intake CD > AEGL-2	–
Category 2 Intermediate Consequence	0.25 Sv (25 rem) <RD≤ 1 Sv (100 rem) For the worker (elsewhere in room), except the worker (local), AEGL-2 < CD≤ AEGL-3 For the worker (local), AEGL-2 < CD ≤ AEGL-3 for HF ** < CD ≤ * for U	0.05 Sv (5 rem) < RD≤ 0.25 Sv (25 rem) AEGL-1 <CD≤ AEGL-2	Radioactive release > 5000 x Table 2 Appendix B of 10 CFR Part 20
Category 1 Low Consequence	Accidents of lower radiological and chemical exposures than those above in this column	Accidents of lower radiological and chemical exposures than those above in this column	Radioactive releases with lower effects than those referenced above in this column

Notes:

*NUREG-1391 threshold value for intake of soluble U resulting in permanent renal failure

**NUREG-1391 threshold value for intake of soluble U resulting in no significant acute effects to an exposed individual

ERPG and AEGL Values For HF (values in mg HF/m^3)

ERPG		AEGL					
	1-hr		**10-min**	**30-min**	**1-hr**	**4-hr**	**8-hr**
ERPG-1	1.6	**AEGL-1**	0.8	0.8	0.8	0.8	0.8
ERPG-2	16.4	**AEGL-2**	78	28	20	9.8	9.8
ERPG-3	41	**AEGL-3**	139	51	36	18	18

Table 6.3-4 ERPG and AEGL values for Uranium Hexafluoride (as soluble U)
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ERPG and AEGL Values For UF_6 (values in mg soluble U/m^3)

ERPG		AEGL					
	1-hr		**10-min**	**30-min**	**1-hr**	**4-hr**	**8-hr**
ERPG-1	3.4	**AEGL-1**	2.4	2.4	2.4	NR	NR
ERPG-2	10	**AEGL-2**	19	13	6.5	1.6	0.8
ERPG-3	20	**AEGL-3**	146	49	24	6.1	3.1

Table 6.3-5 Definition of Consequence Severity Categories

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		High Consequence (Category 3)	Intermediate Consequence (Category 2)
Acute Radiological Doses	Worker	>100 rem TEDE	>25 rem TEDE
	Outside Controlled Area	>25 rem TEDE	>5 rem TEDE
Acute Radiological Exposure	Worker	not applicable	not applicable
	Outside Controlled Area	>30 mg U intake	>5.4 mg U/m^3 (24-hr average)
Acute Chemical Exposure	Worker (local)	>40 mg U intake; > 139 mg HF/m^3	>10 mg U intake; >78 mg HF/m^3
	Worker (elsewhere in room)	>146 mg U/m^3; > 139 mg HF/m^3	>19 mg U/m^3; >78 mg HF/m^3
	Outside Controlled Area (30-min exposure)	>13 mg U/m^3; >28 mg HF/m^3	>2.4 mg U/m^3; >0.8 mg HF/m^3

Table 6.3-6 Health Effects from Intake of Soluble Uranium
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Health Effects	Uranium Intake (mg) by 70 kg Person
50% Lethality	230
Threshold for Intake Resulting in Permanent Renal Damage	40
Threshold for Intake Resulting in No Significant Acute Effects	10
No Effect	4.3

FIGURES

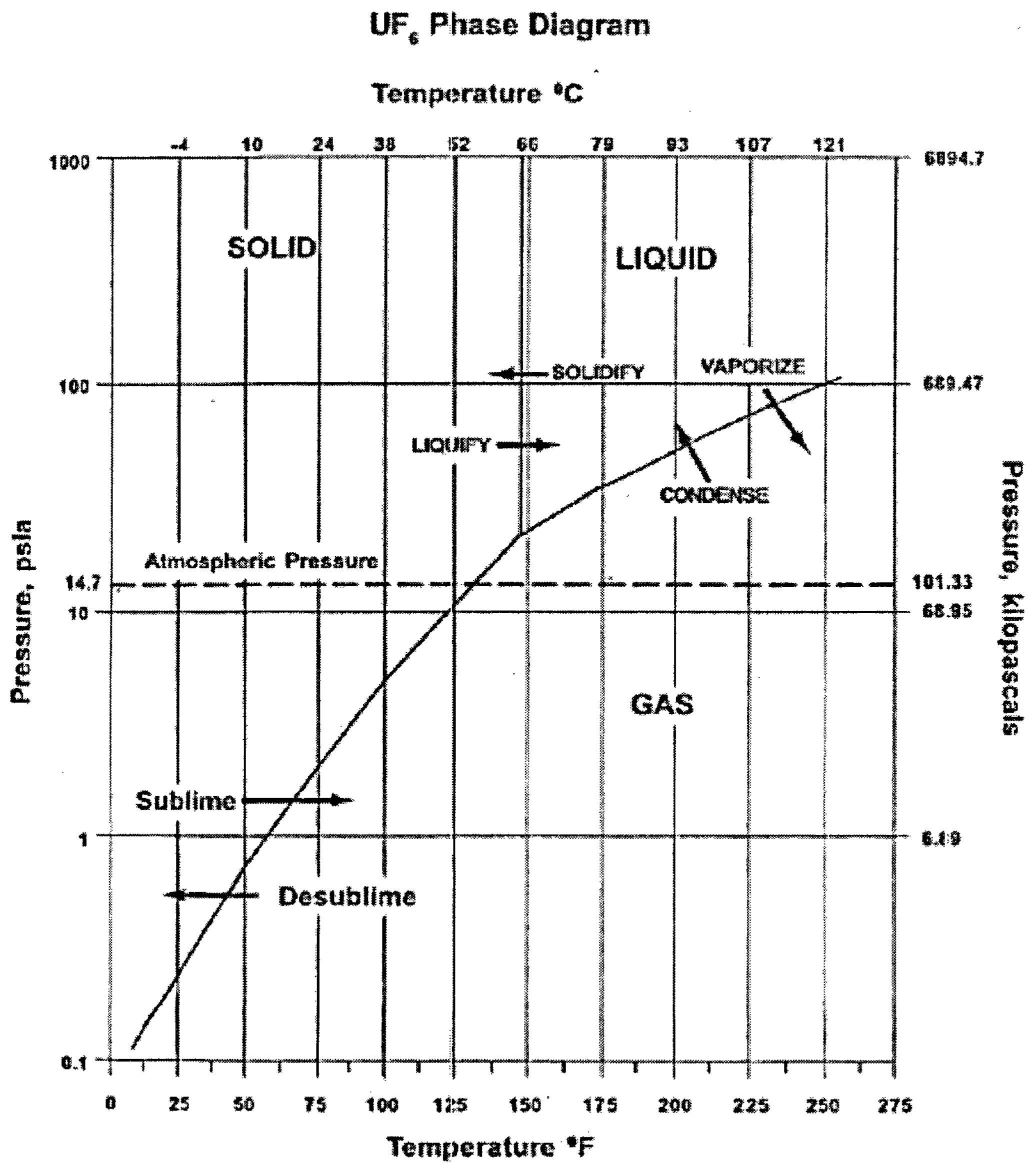

FIGURE 6.1-1

UF_6 PHASE DIAGRAM

REVISION DATE: DECEMBER 2003

REFERENCE NUMBER
Figures 6.1.doc

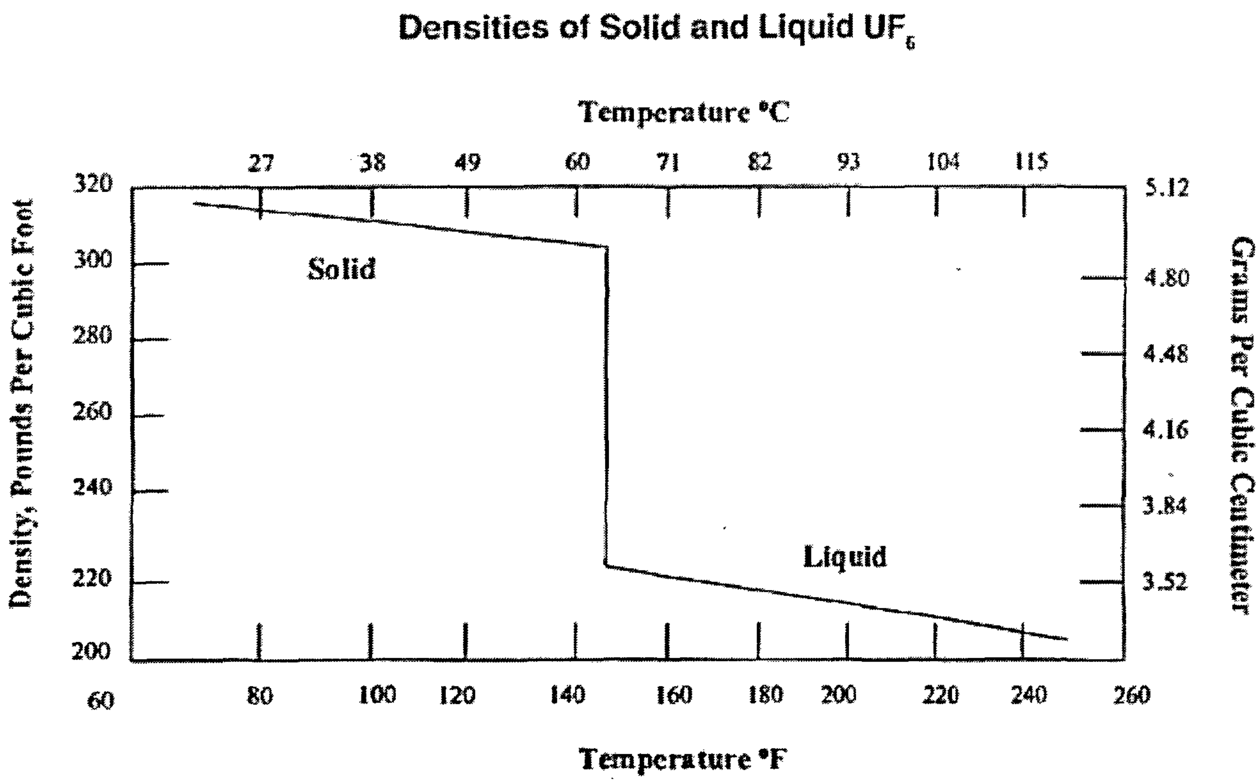

FIGURE 6.1-2

DENSITIES OF SOLID AND LIQUID UF_6

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7.0 FIRE SAFETY

This chapter documents the National Enrichment Facility (NEF) fire safety program. The fire safety program is part of the overall facility safety program and is intended to reduce the risk of fires and explosions at the facility. The facility safety program is described in Chapter 3, Integrated Safety Analysis Summary. The fire safety program documents how the facility administers and ensures fire safety at the facility.

The NEF fire safety program meets the acceptance criteria in Chapter 7 of NUREG-1520 (NRC, 2002) and is developed, implemented and maintained in accordance with the requirements of 10 CFR 70.62(a) (CFR, 2003a), 10 CFR 70.22 (CFR, 2003b) and 10 CFR 70.65 (CFR, 2003c). In addition, the fire safety program complies with 10 CFR 70.61 (CFR, 2003d), 10 CFR 70.62 (CFR, 2003a) and 10 CFR 70.64 (CFR, 2003e). NUREG/CR-6410 (NRC, 1998), NUREG-1513 (NRC, 2001) NRC Generic Letter 95-01 (NRC, 1995) and NFPA 801 (NFPA, 2003) were utilized as guidance in developing this chapter.

The information provided in this chapter, the corresponding regulatory requirement and the section of NUREG-1520 (NRC, 2002), Chapter 7 in which the Nuclear Regulatory Commission (NRC) acceptance criteria are presented is summarized below:

Information Category and Requirement	10 CFR 70 Citation	NUREG-1520 Chapter 7 Reference
Section 7.1 Fire Safety Management Measures	70.62(a), (d) & 70.64(b)	7.4.3.1
Section 7.2 Fire Hazards Analysis	70.61(b), (c) & 70.62(a)&(c)	7.4.3.2
Section 7.3 Facility Design	70.62(a), (c) & 70.64(b)	7.4.3.3
Section 7.4 Process Fire Safety	70.64(b) & 70.64(b)	7.4.3.4
Section 7.5 Fire Protection and Emergency Response	70.62(a), (c) & 70.64(b)	7.4.3.5

7.1 FIRE SAFETY MANAGEMENT MEASURES

Fire safety management measures establish the fire protection policies for the site. The objectives of the fire safety program are to prevent fires from starting and to detect, control, and extinguish those fires that do occur. The fire protection organization and fire protection systems at the NEF provide protection against fires and explosions based on the structures, systems, and components (SSC) and defense-in-depth practices described in this chapter. Fire barriers and administrative controls are considered fire protection items relied on for safety (IROFS).

7.1.1 Fire Protection IROFS

IROFS associated with fire protection are specified in Section 3.8, Items Relied on for Safety (IROFS).

7.1.2 Management Policy and Direction

Louisiana Energy Services (LES) is committed to ensuring that the IROFS, as identified in the ISA Summary, are available and reliable, and that the facility maintains fire safety awareness among employees, controls transient ignition sources and combustibles, and maintains a readiness to extinguish or limit the consequences of fire. The facility maintains fire safety awareness among employees through its General Employee Training Program. The training program is described in Chapter 11, Management Measures.

The responsibility for fire protection rests with the Health, Safety & Environment (HS&E) Manager who reports directly to the Plant Manager. The HS&E Manager is assisted by the Industrial Safety Manager, whose direct responsibility is to ensure the day-to-day safe operation of the facility in accordance with occupational safety and health regulations, including the fire safety program. Fire protection engineering support is provided by the engineering manager in Technical Services. The personnel qualification requirements for the HS&E Manager and the Industrial Safety Manager are presented in Chapter 2, Organization and Administration.

The Industrial Safety Manager is assisted by fire safety personnel who are trained in the field of fire protection and have practical day-to-day fire safety experience at nuclear facilities. The fire protection staff is responsible for the following:

- Fire protection program and procedural requirements
- Fire safety considerations
- Maintenance, surveillance, and quality of the facility fire protection features
- Control of design changes as they relate to fire protection
- Documentation and record keeping as they relate to fire protection
- Fire prevention activities (i.e., administrative controls and training)
- Organization and training of the fire brigade
- Pre-fire planning.

The facility maintains a Safety Review Committee (SRC) that reports to the Plant Manager. The SRC performs the function of a fire safety review committee. The SRC provides technical and administrative review and audit of plant operations including facility modifications to ensure that fire safety concerns are addressed.

Engineering review of the fire safety program is accomplished by configuration management and the SRC. Configuration management is discussed in Chapter 11, Management Measures, and the SRC is discussed in Chapter 2, Organization and Administration.

The subject matter discussed in Section 7.1.2 is essentially the same as the subject matter discussed in the Claiborne Enrichment Center Safety Analysis Report (LES, 1993). The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) relative to Management Policy and Direction (Program Management) and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion on Management Policy and Direction (Program Management) is discussed in NUREG –1491 (NRC, 1994), Section 4.6.

7.1.3 Fire Prevention

Administrative controls are used to maintain the performance of the fire protection systems and delineate the responsibilities of personnel with respect to fire safety. The primary fire safety administrative controls are those that relate to fire prevention. These fire prevention controls, in the form of procedures, primarily control the storage and use of combustible materials and the use of ignition sources. These controls include, but are not limited to, the following:

- Governing the handling of transient combustibles in buildings containing IROFS, including work-generated combustibles
- Implementing a permit system to control ignition sources that may be introduced by welding, flame cutting, brazing, or soldering operations
- Ensuring that the use of open flames or combustion-generated smoke for leak testing is not permitted
- Conducting formal periodic fire prevention inspections to (1) ensure that transient combustibles adhere to established limits based on the Fire Hazard Analysis; (2) ensure the availability and acceptable condition of fire protection systems/equipment, fire stops, penetration seals, and fire-retardant coatings; and (3) ensure that prompt and effective corrective actions are taken to correct conditions adverse to fire protection and preclude their recurrence
- Performing periodic housekeeping inspections
- Implementing a permit system to control the disarming of fire detection or fire suppression systems, including appropriate compensatory measures
- Implementing fire protection system inspection, testing, and maintenance procedures.

7.1.4 Inspection, Testing, and Maintenance of Fire Protection Systems

An inspection, testing and maintenance program is implemented to ensure that fire protection systems and equipment remain operable and function properly when needed to detect and suppress fire. Fire protection procedures are written to address such topics as training of the fire brigade, reporting of fires, and control of penetration seals. The facility's Industrial Safety group has responsibility for fire protection procedures in general; with the facility's maintenance section having responsibility for certain fire protection procedures such as control of repairs to facility penetration seals. Refer to Chapter 11, Management Measures, for additional information on procedures and maintenance activities.

The subject matter discussed in Section 7.1.4 is essentially the same as the subject matter discussed in the Claiborne Enrichment Center SAR (LES, 1993). The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) relative to Inspection, Testing, and Maintenance of Fire Protection Systems (Fire Protection Equipment Maintenance) and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion on Inspection, Testing, and Maintenance of Fire Protection Systems (Fire Protection Equipment Maintenance) is discussed in NUREG –1491 (NRC, 1994), Section 4.6.

7.1.5 Emergency Organization Qualifications, Drills and Training

The qualifications, drills and training of the fire brigade members who are part of the Emergency Organization are in accordance with NFPA 600 (NFPA, 1996i). The primary purpose of the Fire Brigade Training Program is to develop a group of facility employees trained in fire prevention, fire fighting techniques, first aid procedures, and emergency response. They are trained and equipped to function as a team for the fighting of fires.

The Fire Brigade Program provides entrance and educational requirements for fire brigade candidates as well as the medical- and job-related physical requirements. The Fire Brigade Training Program provides for initial training of all new fire brigade members, semi-annual classroom training and drills, annual practical training, and leadership training for fire brigade leaders.

The NEF Emergency Plan also discusses the use of offsite emergency organizations, drills and training.

7.1.6 Pre-Fire Plans

Detailed pre-fire plans will be developed for use by the facility fire brigade.

The pre-fire plans include the location of fire protection equipment, approach paths for fire response, potential hazards in the area, power supply and ventilation isolation means, important plant equipment in the area and other information considered necessary by fire emergency response personnel.

The subject matter discussed in Section 7.1.6 is essentially the same as the subject matter discussed in the Claiborne Enrichment Center SAR (LES, 1993). The NRC staff previously reviewed the Claiborne Enrichment Center SAR (LES, 1993) relative to Pre-Fire Plans and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion on Pre-Fire Plans is discussed in NUREG –1491 (NRC, 1994), Section 4.6.

7.2 FIRE HAZARDS ANALYSIS

A Fire Hazards Analysis (FHA) has been conducted for the facility including the fire areas and fire zones which if uncontrolled, could release UF_6 in quantity and form that could cause an intermediate or high consequence, as defined in 10 CFR 70.61 (CFR, 2003d). UF_6 is present in the Technical Services Building (TSB), Blending and Liquid Sampling Area, UF_6 Handling Area, Separations Building, Cylinder Receipt and Dispatch Building (CRDB), Centrifuge Test and Post Mortem Facilities in the Centrifuge Assembly Building (CAB) and the UBC Storage Pad.

The FHA develops bounding credible fire scenarios and then assesses the consequences of unmitigated fire.

The FHA for the facility consists of the following:

- A description of the facility's use and function
- The specific fire hazards and potential fire scenarios within the fire areas and fire zones
- The methods of consequence analysis
- The occupancy and construction requirements
- Life safety requirements
- The boundaries of the fire areas and fire zones
- The IROFS affected by the postulated fire scenarios within the fire area
- The facility response to the postulated fires
- Defense or mitigation strategy for overall facility protection.

The results of the FHA are utilized in the Integrated Safety Analysis (ISA) to identify possible fire initiators and accident sequences leading to radiological consequences or toxic chemical consequences resulting from interaction with UF_6.

The FHA is updated and controlled by configuration management as discussed in Chapter 11, Management Measures, to ensure that the information and analysis presented in the FHA are consistent with the current state of the facility. The FHA is reviewed and updated as necessary to incorporate significant changes and modifications to the facility, its processes, or combustible inventories.

7.3 FACILITY DESIGN

The design of the facility incorporates the following:

- Limits on areas and equipment subject to contamination
- Design of facilities, equipment, and utilities to facilitate decontamination.

7.3.1 Building Construction

The facility consists of several different buildings or functional areas:

- Visitor Center
- Site Security Buildings
- Administration Building
- Technical Services Building (TSB)
- Central Utilities Building (CUB).
- Separations Building (consisting of three Separations Building Modules), which include:

UF_6 Handling Area
Cascade Halls
Process Services Area.

- Cylinder Receipt and Dispatch Building (CRDB)
- Blending and Liquid Sampling Area
- Centrifuge Assembly Building (CAB)
- Centrifuge Test and Centrifuge Post Mortem Facilities (within the CAB)
- UBC Storage Pad
- Fire Water Pump Building.

The Visitor Center, Security Buildings, Administration Building, Fire Water Pump Building and Tanks and CUB are independent of the rest of the plant main buildings. The Visitor Center is located outside of the Controlled Area security fence. The Administration Building, Fire Water Pump Building and the CUB are provided with automatic sprinkler protection. The remaining buildings/areas have no automatic sprinkler protection.

The TSB, Separations Building, CRDB, Blending and Liquid Sampling Area, CAB and Centrifuge Test and Centrifuge Post Mortem Area are pre-cast concrete frame and concrete panel construction with an upside down ballasted roof system over pre-cast concrete tees. This construction is classified as Type I, Unsprinklered in accordance with the New Mexico Building Code (NMBC) (NMBC, 1997) and as Type I Construction by NFPA 220 (NFPA, 1999). The Administration Building, Fire Water Pump Building and the CUB are unprotected steel frame buildings with insulated metal panel exterior walls and with built-up roofing on metal deck roof. This construction is classified as Type III N, Unprotected, Sprinklered in accordance with the NMBC (NMBC, 1997) and as Type II Construction by NFPA 220 (NFPA, 1999). The Visitor

Center and the Site Security Buildings are unprotected steel frame buildings with insulated metal panel exterior walls and with built-up roofing on metal deck roof. This construction is classified as Type III N, Unprotected, in accordance with the NMBC (NMBC, 1997) and as Type II Construction by NFPA 220 (NFPA, 1999).

The UBC Storage Pad is an open lay-down area and consists of a concrete pad with a dedicated collection and drainage system. Concrete saddles are used for storage of cylinders approximately 200 mm (8 in) above ground level. There is no building for the UBC Storage Pad.

7.3.2 Fire Area Determination and Fire Barriers

The facility is subdivided into fire areas by barriers with fire resistance commensurate with the potential fire severity, in accordance with NFPA 101 (NFPA, 1997a) and the NMBC (NMBC, 1997). The design and construction of fire barrier walls is in accordance with NFPA 221 (NFPA, 1997b). These fire areas are provided to limit the spread of fire, protect personnel and limit the consequential damage to the facility. Fire barriers are shown in Figures 7.3-1 through 7.3-8. The fire resistance rating of fire barrier assemblies is determined through testing in accordance with NFPA 251 (NFPA, 1995d). Openings in fire barriers are protected consistent with the designated fire resistance rating of the barrier. Penetration seals provided for electrical and mechanical openings are listed to meet the guidance of ASTM E-814 (ASTM, 2002) or UL 1479 (UL, 2003). Penetration openings for ventilation systems are protected by fire dampers having a rating equivalent to that of the barrier. Door openings in fire rated barriers are protected with fire rated doors, frames and hardware in accordance with NFPA 80 (NFPA, 1995b).

7.3.3 Electrical Installation

All electrical systems at the facility are installed in accordance with NFPA 70 (NFPA, 1996e). Switchgear, motor control centers, panel boards, variable frequency drives, uninterruptible power supply systems and control panels are mounted in metallic enclosures and contain only small amounts of combustible material. Cable trays and conduits are metallic and the cables in cable trays are flame retardant and tested in accordance with the guidance of ANSI / IEEE 383 (ANSI / IEEE, 1974), IEEE 1202 (IEEE, 1991), UL 1277 (UL, 2001), and ICEA T-29-520 (ICEA, 1986).

Lighting fixtures are constructed of non-combustible materials and their ballasts are electronic and contain only an insignificant amount of combustible material.

All indoor transformers are dry type. Outdoor oil filled transformers are provided by the local utility and are located in the local utilities substation yard which is located at the southwest corner of the facility with adequate spatial separation from the facility buildings so as not to present an exposure fire hazard to the facility.

An auxiliary power system is provided to supply power for temporary lighting, ventilation and radiation-monitoring equipment where potential radiation hazard exists.

Electrical conduits leading to or from areas with uranic material are sealed internally to prevent the spread of radioactive materials. Only utilities required for operation within areas having uranic material enter into these areas.

7.3.4 Life Safety

The buildings are provided with means of egress, illumination, and protection in accordance with NFPA 101 (NFPA, 1997a). Barriers with fire resistance ratings consistent with NFPA 101 (NFPA, 1997a) and the FHA are provided to prevent unacceptable fire propagation.

All of the buildings are provided with emergency lighting for the illumination of the primary exit paths and in critical operations areas where personnel are required to operate valves, dampers and other controls in an emergency. Emergency lighting is considered as a critical load. All critical loads are fed from the uninterruptible power supplies (UPSs) which are connected to the essential load motor control centers (MCCs). The UPSs receive power input from two incoming power sources, two diesel powered electric generators and stationary batteries. All power inputs to the UPS transfer automatically to another source if the first source fails. Thus, loads connected to the UPS are unaffected by offsite power and standby generator failure.

Marking of means of egress, including illuminated exit signs, are provided in accordance with NFPA 101 (NFPA, 1997a) Section 5.10 and Chapter 10 of the NMBC (NMBC, 1997).

7.3.5 Ventilation

The building heating, ventilating and air conditioning (HVAC) system provides the primary form of ventilation employed at the facility. The HVAC system is designed to maintain room temperature and the specific environmental conditions associated with processes undertaken within a particular area. The TSB HVAC System also performs a confinement ventilation function to effectively reduce the potential chronic exposure of individuals working at the plant and to the public, to hazardous materials.

The ventilation system is not engineered for smoke control. It is designed to shutdown in the event of a fire. Ductwork, accessories and support systems are designed and tested in accordance with NFPA 801 (NFPA, 2003), NFPA 90A (NFPA, 1996g), NFPA 90B (NFPA, 1996h), and NFPA 91 (NFPA, 1995c). Flexible air duct couplings in ventilation and filter systems are noncombustible. Air entry filters are UL Class I.

The power supply and controls for mechanical ventilation systems are located outside the fire area served. The ventilation system is designed such that the areas containing dispersible radioactive materials remain at a lower pressure than that of adjoining areas of the facility. These areas include the Mass Spectrometry Laboratory, the Chemical Laboratory, the Ventilated Room, the Cylinder Preparation Room and the Decontamination Workshop. Ductwork from areas containing radioactive materials that pass through non-radioactive areas are constructed of non-combustible material and are protected from possible exposure to fire by materials having an appropriate fire resistance rating.

High efficiency particulate air (HEPA) filtration systems are utilized in various areas in the plant in the confinement ventilation function of the TSB HVAC System, the gaseous effluent vent systems (GEVS) and in the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System. HEPA filters are UL 586 (UL, 1996)(UL Class I), which are non-combustible. In the GEVS and, the Centrifuge Test and Post Mortem Exhaust Filtration System, and the Confinement Ventilation function of the TSB HVAC System, the HEPA filters are enclosed in ductwork. The HEPA filtration systems are analyzed in the FHA. They are designed to shutdown in the event of a fire.

Smoke control is accomplished by the Fire Brigade and off-site Fire Department utilizing portable smoke removal equipment.

7.3.6 Drainage

Water that may discharge from the fire water system or from fire fighting activities could be contaminated with radioactive materials. The water will be contained, stored, sampled, and treated if necessary. This also applies to areas containing flammable and combustible liquids. Wall and floor interfaces will be made watertight. Provisions will be made at all pertinent door openings to prevent fire protection water from migrating outside of the contained area. If there is a possibility that the water could be contaminated with fissile uranium compounds, the containment methodology will be designed to be safe with respect to criticality. The drainage system design and associated containment configuration will be addressed during the design phase and the Safety Analysis Report will be revised, as appropriate. Water runoff from the UBC Storage Pad will be collected in the UBC Storage Pad Stormwater Retention Basin. Liquid effluent monitoring associated with the UBC Storage Pad Stormwater Retention Basin is discussed in the Environmental Report.

7.3.7 Lightning Protection

Lightning protection for the facility is in accordance with NFPA 780 (NFPA, 1997c).

7.3.8 Criticality Concerns

Criticality controls will be provided by employing the basic principals of criticality safety. The premise of nuclear criticality prevention is that at least two, unlikely, independent, and concurrent changes in process conditions must occur before a criticality accident is possible. This double contingency principal is described in ANSI/ANS-8.1-1998 (ANSI, 1998). Controls or systems of controls are used to limit process variables in order to maintain safe operating conditions.

Moderation control is applied for criticality safety of UF_6 at this facility. Neither automatic sprinkler nor standpipe and hose systems are provided in the TSB, Separation Buildings, Blending and Liquid Sampling, CRDB, CAB, and Centrifuge Post Mortem areas. Procedures and training for both onsite fire brigade and offsite fire department emphasize the need for moderator control in these areas.

Fire protection concerns are addressed in the moderation control areas by fire protection IROFS. The IROFS define administrative controls which limit the transient and in situ combustibles, the ignition sources in these areas and isolate these areas from other areas of the plant with appropriately rated fire barriers to preclude fire propagation to or from these areas. There are automatic detection and manual alarm systems located in these areas. Fires will be extinguished in these areas by the fire brigade and / or local fire department with the use of portable and wheeled fire extinguishers. In the unlikely event that extinguisher cannot control or extinguish the fire, then the fire brigade, local fire department and the Emergency Operations Center will work together to ensure that moderator control is maintained in these areas. If deemed appropriate, hose streams are available from fire hydrants located throughout the facility.

See Chapter 5, Nuclear Criticality Safety, for additional discussion on criticality control.

7.3.9 Hydrogen Control

Hydrogen is utilized within the Technical Services Building Chemical Laboratory. In order to prevent the possibility of fire or explosion in the laboratory, areas where hydrogen might accumulate will be protected by one or a combination of following features:

- Hydrogen piping will be provided with excess flow control.
- Hydrogen supply will be isolated by emergency shutoff valves interlocked with hydrogen detection in the area(s) served by the hydrogen piping.
- Natural or mechanical ventilation will be provided to ensure that hydrogen concentrations do not exceed 25% of the lower explosive limit. If mechanical ventilation is provided, it will be continuous or will be interlocked to start upon the detection of hydrogen in the area. Mechanical ventilation will also be provided with airflow sensors to sound an alarm if the fan becomes inoperative.

Hydrogen may also be generated at battery charging stations in the facility. In order to prevent the possibility of explosion or fire, areas where hydrogen might accumulate will be protected by a design which incorporates the following measures, as necessary, that are identified in NFPA 70E (NFPA, applicable version) and/or ANSI C2 (ANSI, applicable version).

- Natural or mechanical ventilation will be provided to ensure that hydrogen concentrations do not exceed 25% of the lower explosive limit. If mechanical ventilation is provided, it will be continuous or will be interlocked to start upon the detection of hydrogen in the area. Mechanical ventilation will also be provided with airflow sensors to sound an alarm if the fan becomes inoperative.

7.3.10 Environmental Concerns

Radiological and chemical monitoring and sampling will be performed as specified in NEF Environmental Report, Chapter 6, Environmental Measurements and Monitoring Programs, on the contaminated and potentially contaminated facility liquid effluent discharge including water used for fire fighting purposes. Discharges from the Liquid Effluent Collection and Treatment System will be routed to the Treated Effluent Evaporative Basin. Surface water runoff will be diverted into water collection basins. Water runoff from the UBC Storage Pad will be collected in the UBC Storage Pad Stormwater Retention Basin. Water runoff from the remaining portions of the site will be collected in the Site Stormwater Detention Basin.

7.3.11 Physical Security Concerns

In no cases will security requirements prevent safe means of egress as required by the NFPA 101 (NFPA, 1997a) and the NMBC (NMBC, 1997).

The Physical Security Plan (PSP) addresses the establishment of permanent and temporary Controlled Areas. The PSP identifies the ingress and egress methodology during both normal and emergency conditions. This includes emergency response personnel both onsite and offsite. Two means of access to the site are provided, one via one of the two controlled gates continuously manned by Security and the other via designated emergency access gates (i.e., crash gates). Refer to the PSP for additional details.

7.3.12 Baseline Design Criteria and Defense-In-Depth

The FHA and the ISA demonstrate that the design and construction of the facility complies with the baseline design criteria (BDC) of 10 CFR 70.64(a) (CFR, 2003e), the defense-in-depth requirements of 10 CFR 70.64(b) (CFR, 2003e) and are consistent with the guidance provided in NFPA 801 (NFPA, 2003). The design provides for adequate protection against fire and explosion by incorporating defense-in-depth concepts such that health and safety are not wholly dependent on any single element of the design, construction, maintenance or operation of the facility. This is accomplished by achieving a balance between preventing fires from starting, quickly detecting, controlling and promptly extinguishing those fires that do occur and protecting structures, systems and components such that a fire that is not promptly extinguished or suppressed will not lead to an unacceptable consequence.

7.4 PROCESS FIRE SAFETY

Chapter 6, Chemical Process Safety, describes the chemical classification process, the hazards of chemicals, chemical process interactions affecting licensed material and/or hazardous chemicals produced from licensed material, the methodology for evaluating hazardous chemical consequences, and chemical safety assurance. The only process chemical of concern is uranium hexafluoride (UF_6). UF_6 is not flammable and does not disassociate to flammable constituents under conditions at which it will be handled at the NEF. The two byproducts in the event of a UF_6 release are hydrogen fluoride (HF) and uranyl fluoride (UO_2F_2) and neither presents a process fire safety hazard. The Integrated Safety Analysis has analyzed the hazards associated with the processes performed at the facility. The analysis did not identify any processes which represented a process fire safety hazard.

7.5 FIRE PROTECTION AND EMERGENCY RESPONSE

This section documents the fire protection systems and fire emergency response organizations provided for the facility.

7.5.1 Fire Protection System

The facility fire protection systems consist of a dedicated fire water supply and distribution system, automatic suppression systems (sprinklers and alternate systems), standpipe and hose systems, portable fire extinguishers, fire detection and alarm systems, fire pump control systems, valve position supervision, system maintenance and testing, fire prevention program, fire department/fire brigade response and pre-fire plans.

7.5.1.1 Fire Water Supply and Distribution System

A single Fire Protection Water Supply System provides storage and distribution of water to the Fire Protection System that protects the entire facility as shown in Figure 7.5-1, Exterior Fire Protection System Overall Site Plan, and Figure 7.5-2, Sprinkler System Coverage.

7.5.1.1.1 System Description

A reliable fire protection water supply and distribution system of adequate flow, pressure, and duration is provided based on the characteristics of the site and the FHA. The fire protection water supply and distribution system is based on the largest fixed fire suppression system demand, including a hose stream allowance, in accordance with NFPA 13 (NFPA, 1996a). The fire protection water supply consists of two 946,354-lL (250,000-gal) (minimum) water storage tanks designed and constructed in accordance with NFPA 22 (NFPA, 1996d). The tanks are used for both fire protection water supply and process water supply. A reserve quantity of 473,179 L (125,000 gal) is maintained in the bottom of each tank for fire protection purposes. The elevation of the suction line for the process water pump is above the level of the required fire protection water supply in each tank. Thus the process water pump cannot pump water required for fire protection purposes. The fire protection water supply in each tank is sized for the maximum anticipated water supply needed to control and extinguish the design basis fire at the facility. Two, 3785 l/min at 10.35 bar (1000 gpm at 150 psi) horizontal, centrifugal, fire pumps designed and installed in accordance with NFPA 20 (NFPA, 1996c) are provided. For redundancy the capacity of the fire protection water supply is designed to ensure that 100% of the required flow rate and pressure are available in the event of failure of one of the water storage tanks or fire pumps. The maximum demand anticipated based on a design basis fire is 3785 l/min (1000 gpm) based on 1982 l/min (500 gpm) flowing from a building sprinkler system plus 1982 l/min (500 gpm) for hose streams for a duration of two hours. The tanks are arranged so that one will be available for suction at all times.

Fill and make up water for the storage tanks are from the city water supply to the site which is capable of filling either storage tank in an 8-hour period.

The fire water service main for the plant is designed and installed in accordance with NFPA 24 (NFPA, 1995a). The distribution system, including piping associated with the fire pumps is looped and arranged so that a single pipe break or valve failure will not totally impair the system per the Fire Hazard Analysis and NFPA 801 (NFPA, 2003). Through appropriate valve alignment, either fire pump can take suction from either storage tank and discharge through either leg of the underground piping loop. The system piping is sized so that the largest sprinkler system demand (including hose stream allowance) is met with the hydraulically shortest flow path assumed to be out of service. Sectional control valves are arranged to provide adequate sectional control of the fire main loop to minimize protection impairments. All fire protection water system control valves are monitored under a periodic inspection program and their proper positioning is supervised in accordance with NFPA 801 (NFPA, 2003). Exterior fire hydrants, equipped with separate shut-off valves on the branch connection, are provided at intervals to ensure complete coverage of all facility structures, including the UBC Storage Pad.

The fire pumps are separated from each other by fire-rated barrier construction. Both pumps are diesel engine-driven. Each pump is equipped with a dedicated listed controller. The pumps are arranged for automatic start functions upon a drop in the system water pressure as detected by pressure switches contained within the pump controllers. Use of start delay timers prevents simultaneous start of both pumps. Each fire pump controller interfaces with the site-wide protective signaling system for all alarm and trouble conditions recommended by NFPA 20 (NFPA, 1996c), which are monitored and annunciated at the central alarm panel in the Control Room. Once activated, the fire pumps can only be shut-off at the pump controller location. Pumps, suction and discharge piping and valves are all provided and arranged in accordance with the recommendations of NFPA 20 (NFPA, 1996c). Dedicated diesel fuel tanks are provided for each pump. These tanks are located in the Fire Water Pump Building and are sized to provide a minimum eight hour supply of fuel in accordance with the recommendations of NFPA 20 (NFPA, 1996c). The Fire Water Pump Building is provided with automatic sprinkler protection.

A jockey pump is provided in the Fire Water Pump Building to maintain pressure in the fire protection system during normal operation.

7.5.1.1.2 System Interfaces

The Fire Protection Water Supply System interfaces with the city water supply that supplies fill and make up water to the fire water supply storage tanks.

7.5.1.1.3 Safety Considerations

Failure of the Fire Water Supply and Distribution System will not endanger public health and safety. The system is designed to assure water supply to automatic fire protection systems, standpipe systems and to fire hydrants located around the facility. This is accomplished by providing redundant water storage tanks and redundant fire pumps which are not subject to a common failure, electrical or mechanical.

7.5.1.2 Standpipe and Hose Systems

As required by the FHA, standpipe systems and interior fire hose stations are provided and installed in accordance with NFPA 14 (NFPA, 1996b) in the following locations:

- Class II standpipe systems for fire brigade and the offsite fire department use are provided in the CUB, CAB and the Administration Building.

- Standpipes and fire hose stations are positioned so that any interior location in the CUB, CAB and the Administration Building can be protected with an effective hose stream.

Each fire hose station is equipped with 30.5 m (100 ft) of 38 mm (1½-in) fire hose and the type of hose nozzle suitable for the hazard protected. The systems are designed to provide a minimum flow recommended by NFPA 14 (NFPA, 1996b) for class II standpipe systems. The systems are separated from the building sprinkler system. The separation ensures that a single impairment will not disable both the sprinklers and the hose systems.

In addition to fixed standpipes and fire hose stations, the NEF will be provided with fire hose on mobile apparatus and/or at strategic locations throughout the facility. The amount of hose provided will be sufficient to ensure that all points within the facility will be able to be reached by at least two 38 mm (1½-in) diameter attack hose lines and one 64 mm (2½-in) diameter backup hose line consistent with NFPA 1410 (NFPA, 2000). These lines are intended for use by the offsite fire response agencies in the event of a structural fire. Hydraulic margin for these hose lines will be sufficient to ensure minimum nozzle pressures of 4.5 bar (65 psia) for attack hose line(s) and 6.9 bar (100 psia) for the backup hose line.

7.5.1.3 Portable Extinguishers

Portable fire extinguishers are installed throughout all buildings in accordance with NFPA 10 (NFPA, 1994). Multi-purpose extinguishers are provided generally for Class A, B, or C fires.

The portable fire extinguishers are spaced within the travel distance limitation and provide the area coverage specified in NFPA 10 (NFPA, 1994). Specialized extinguishers are located in areas requiring protection of particular hazards. Wheeled extinguishers are provided for use in water exclusion areas.

In areas with moderator control issues, the chemical fill for the extinguishers is carbon dioxide and dry chemical and has been selected so as not to create an uncontrolled moderator source.

7.5.1.4 Automatic Suppression Systems

Wet pipe sprinkler systems are engineered to protect specific hazards in accordance with parameters established by the FHA. Water flow detectors are provided to alarm and annunciate sprinkler system actuation. Sprinkler system control valves are monitored under a periodic inspection program and their proper positioning is supervised in accordance with NFPA 801 (NFPA, 2003) to ensure the systems remain operable. The areas of sprinkler system coverage are shown in Figure 7.5-2, Sprinkler System Coverage.

Automatic wet pipe sprinkler systems, designed and tested in accordance with NFPA 13 (NFPA, 1996a), are provided in the following buildings:

- Administration Building
- Central Utilities Building (CUB)
- Fire Pump House.

Fire rated enclosures are provided for several chemical traps located on the second floor of the Process Services Area in each Separations Building Module. These enclosures will be protected with a gaseous suppression system. The particular type of suppression system utilized will be determined in the final design and will be designed and installed in accordance with the applicable NFPA standard, NFPA 12 (NFPA, 1993) for carbon dioxide systems or NFPA 2001 (NFPA, 1996j) for clean agent suppression systems.

7.5.1.5 Fire Detection Systems

All facility structures are provided with automatic fire detectors in accordance with NFPA 72 (NFPA, 1996f) and as required by the FHA. Automatic fire detectors are installed in accordance with NFPA 72 (NFPA, 1996f), NFPA 101 (NFPA, 1997a) and as required by the FHA.

7.5.1.6 Manual Alarm Systems

All facility structures are provided with manual fire alarm pull stations in accordance with NFPA 72 (NFPA, 1996f), NFPA 101 (NFPA, 1997a) and as required by the FHA.

7.5.1.7 Fire Alarm System

Each building of the facility is equipped with a listed, modular, multi-zone fire alarm control panel installed in accordance with NFPA 72 (NFPA, 1996f). Each panel has a dual power supply, consisting of normal building power and backup power by either 24-hour battery or the facility UPS. The method of backup power will be determined in final design. Sprinkler system and hose station water flow detection devices are connected to separate control panel zone modules. Fire detector and manual pull station alarm circuits are also connected to dedicated control panel zone modules. Fire detector zone modules include detector confirmation features to reduce the potential for false alarms. Each zone module has individual disable switches so individual zones can be removed from service for maintenance and trouble shooting without disabling the entire control panel. Each zone module has separate alarm and trouble contacts for connection to the central alarm panel in the Control Room. Activation of a fire detector, manual pull station or water flow detector results in an audible and visual alarm at the building control panel and the central alarm panel.

The central alarm panel, located in the Control Room, is a listed, microprocessor-based addressable console. The central alarm panel has dual power supplies, consisting of normal building power and backup power by either 24-hour battery or the facility UPS. The method of backup power will be determined in final design. The central alarm panel monitors all functions associated with the individual building alarm panels and the fire pump controllers. All alarm and trouble functions are audibly and visually annunciated by the central alarm panel and automatically recorded via printout. Failure of the central alarm panel will not result in failure of any building fire alarm control panel functions.

The following conditions are monitored by the central alarm console through the fire pump controllers:

- Pump running
- Pump failure to start
- Pump controller in "off" or "manual" position
- Battery failure
- Diesel overspeed

- Diesel high engine jacket coolant temperature
- Diesel low oil pressure
- Battery charger failure.

Both pumps are maintained in the automatic start condition at all times, except during periods of maintenance and testing. Remote manual start switches are provided in the Control Room adjacent to the alarm console. Pumps are arranged for manual shut-off at the controllers only.

All fire protection water system control valves are monitored under a periodic inspection program and their proper positioning is supervised in accordance with NFPA 801 (NFPA, 2003).

7.5.2 Fire Emergency Response

7.5.2.1 Fire Brigade

The facility maintains a fire brigade made up of employees trained in fire prevention, fire fighting techniques, first aid procedures, emergency response, and criticality safety. The criticality safety training addresses water moderation, water reflection, product cylinder safety by moderation control, and water flooding. The fire brigade is organized, operated, trained and equipped in accordance with NFPA 600 (NFPA, 1996i). The fire brigade is considered an incipient fire brigade as classified under NFPA 600 (NFPA, 1996i), e.g., not required to wear thermal protective clothing nor self-contained breathing apparatus during firefighting. The intent of the facility fire brigade is to be able to handle all minor fires and to be a first response effort designed to supplement the local fire department for major fires at the plant. The fire brigade members are trained and equipped to respond to fire emergencies and contain fire damage until offsite help from a neighboring fire department arrives. This will include the use of hand portable and wheeled fire extinguishers as well as hoselines to fight interior/exterior incipient fires and to fight larger exterior fires in a defensive mode (e.g., vehicle fires). When the local fire department arrives onsite, the local fire department assumes control and is responsible for all fire fighting activities. The plant fire brigade, working with the plant's Emergency Operations Center, will coordinate offsite fire department activities to ensure moderator control and criticality safety. The fire brigade is staffed so that there are a minimum of five fire brigade members available per shift. The fire brigade includes a safety officer who is responsible to ensure that moderator concerns for criticality safety are considered during firefighting activities.

Periodic training is provided to offsite assistance organization personnel in the facility emergency planning procedures. Facility emergency response personnel meet at least annually with each offsite assistance group to accomplish training and review items of mutual interest including relevant changes to the program. This training includes facility tours, information concerning facility access control (normal and emergency), potential accident scenarios, emergency action levels, notification procedures, exposure guidelines, personnel monitoring devices, communications, contamination control, moderator control issues, and the offsite assistance organization role in responding to an emergency at the facility, as appropriate.

7.5.2.2 Off-Site Organizations

LES will use the services of local, offsite fire departments to supplement the capability of the facility Fire Brigade. The two primary agencies that will be available for this response are the City of Eunice, New Mexico Fire and Rescue Agency and the City of Hobbs, New Mexico Fire

Department. Both of these agencies are signatories to the Lea County, New Mexico Mutual Aid agreement and can request additional mutual aid from any of several county fire departments/fire districts.

A Memorandum of Understanding is in place between LES and these two local fire departments that defines the fire protection and emergency response commitments between the organizations. The training and conduct of emergency drills and the Memoranda of Understanding are discussed in the NEF Emergency Plan.

LES has performed a baseline needs assessment evaluating the response to fires and related emergencies to confirm adequacy of the response considering both facility resources and response of the two primary response agencies. This assessment identified that with some supplemental resource and training development, adequate response is assured.

Eunice Fire and Rescue, as the initial response agency, is comprised of a roster of approximately 20 volunteers. Eunice has three structural fire engines, three grass fire trucks, one water tanker, two command vehicles, and three ambulances, each equipped to provide intermediate level life support. Firefighters are trained to a minimum Firefighter Level I and ambulance personnel to a minimum of Emergency Medical Technician (EMT) – Basic per New Mexico standards.

The Hobbs Fire Department, as the secondary response agency, is comprised of a roster of approximately 70 paid personnel, staffing three fire stations in a three-shift rotation. The department has five structural engines, a ladder truck, a heavy rescue, three grass fire trucks, one water tanker, several command vehicles and six ambulances, each equipped to provide advanced level life support. Firefighters are required to be a minimum Firefighter Level I and EMT – Basic per New Mexico standards. Shift assigned ambulance personnel are EMT – Paramedics per New Mexico standards.

The estimated response time to NEF for a basic life support ambulance is 11 minutes with a second ambulance available within an additional seven minutes. NEF personnel will be trained and equipped to provide first aid and circulatory/respiratory support in the interim (e.g., provide CPR, apply automatic external defibrillation, and administer oxygen).

The estimated response time to NEF for a structural fire engine and full structural crew from Eunice Fire and Rescue is between 11 and 15 minutes. In the event of a fire, the NEF fire brigade will respond and Eunice Fire and Rescue will be notified to respond. If the fire is incipient, the NEF fire brigade will fight the fire utilizing hand portable/wheeled fire extinguishers and/or 38 mm (1½-in) hose lines. In the event that structural fire response is needed, the Hobbs Fire Department will also be notified to respond and the 38 mm (1½-in) and/or 64 mm (2½-in) hose lines from the NEF fire water supply system to the nearest points to the fire will be extended by the NEF fire brigade, where it can be done safely. The latter activity will minimize deployment time for the offsite responders upon their arrival. To ensure that application of water or other firefighting activities are consistent with moderator concerns for criticality safety, the NEF fire brigade safety officer is trained and equipped to don structural firefighting gear and will accompany offsite responders to the firefighting location. In the event that offsite responders are needed in more than one facility location, the criticality safety role of the NEF fire brigade safety officer is fulfilled by appropriately trained NEF personnel (typically fire brigade members). These NEF personnel are trained in criticality safety and trained and equipped to don structural firefighting gear to accompany the offsite responders to required facility locations.

In order to respond to airborne release emergencies or other chemical incidents, NEF will maintain full hazardous material response capability. This is further described in SAR Section 6.4.8, Emergency Planning.

Through a combination of onsite capability, offsite responders, or through contract arrangements, LES will ensure that capabilities are in place to respond to other events such as confined space rescue, trench rescue, high angle rescue, and other technical emergencies as required. The NEF fire brigade/emergency response team equipment will also be inventoried, inspected and tested in accordance with recognized standards. Final needs for these response areas and response equipment will be reassessed after detailed facility design to ensure adequate response capabilities are in place and training completed prior to any construction activities.

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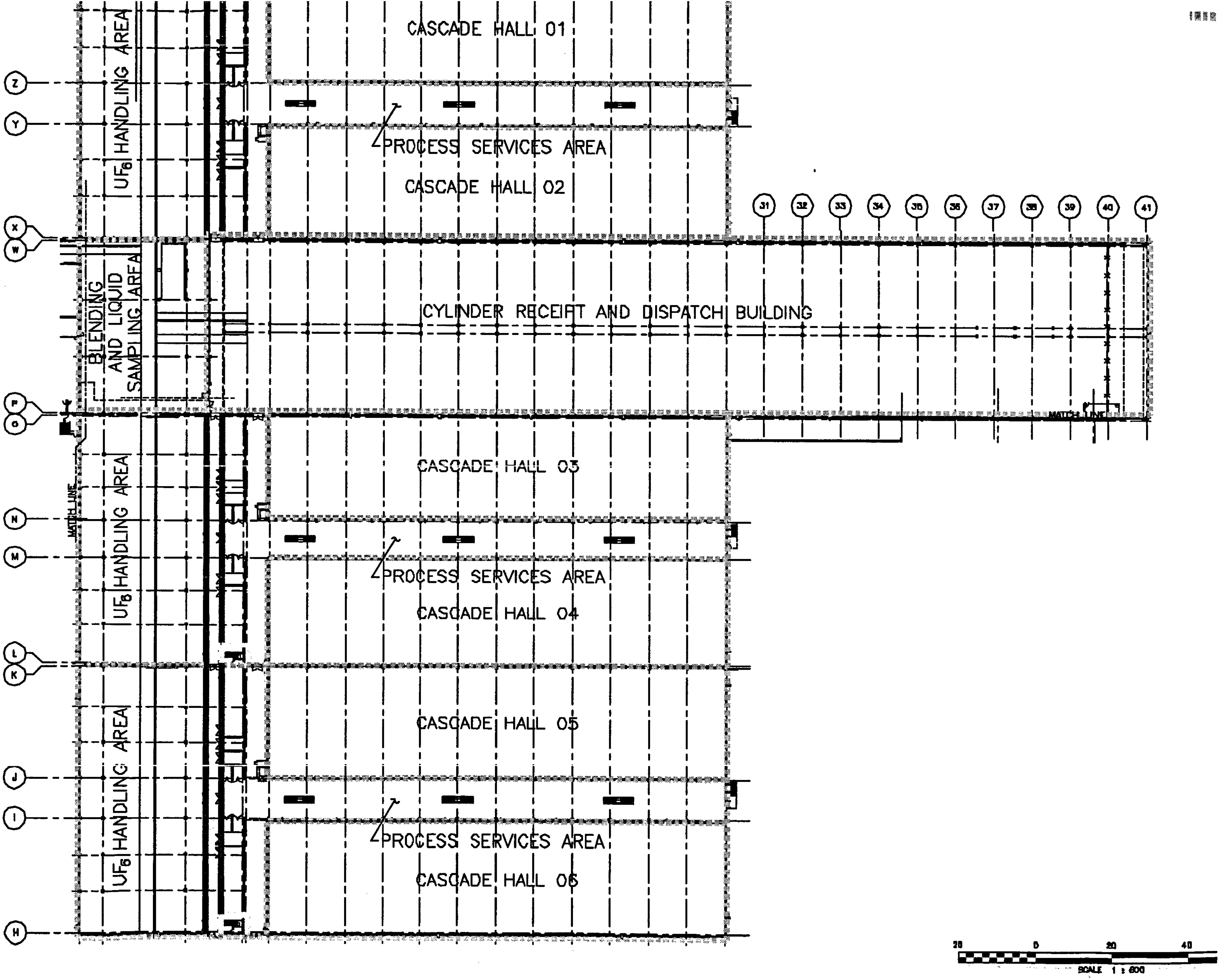
FIGURES



CASCADE HALL 01
PROCESS SERVICES AREA
CASCADE HALL 02
UF6 HANDLING AREA
BLENDING AND LIQUID SAMPLING AREA
CYLINDER RECEIPT AND DISPATCH BUILDING
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CASCADE HALL 03
PROCESS SERVICES AREA
CASCADE HALL 04
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UF6 HANDLING AREA
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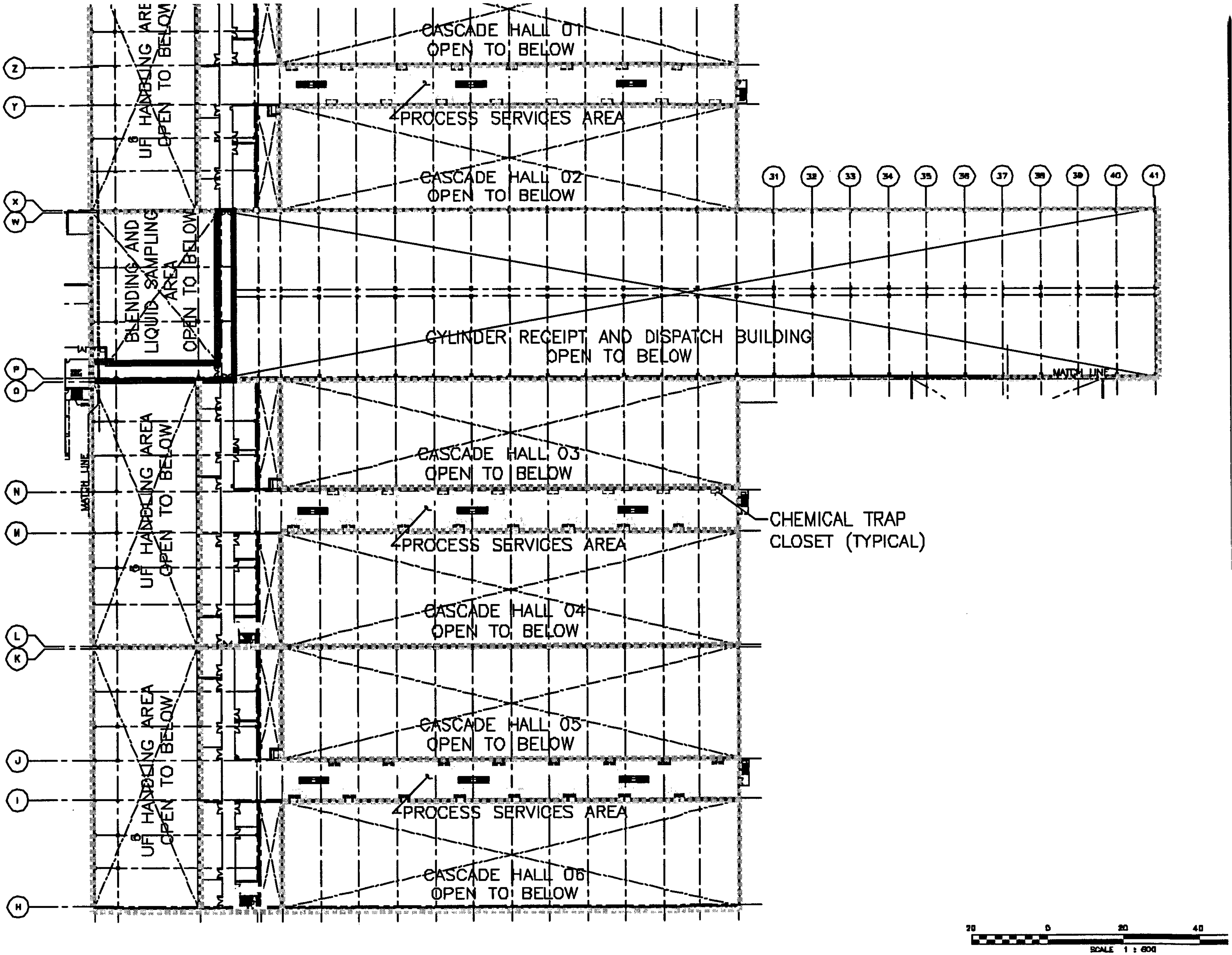

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BLENDING AND
LIQUID SAMPLING
AREA
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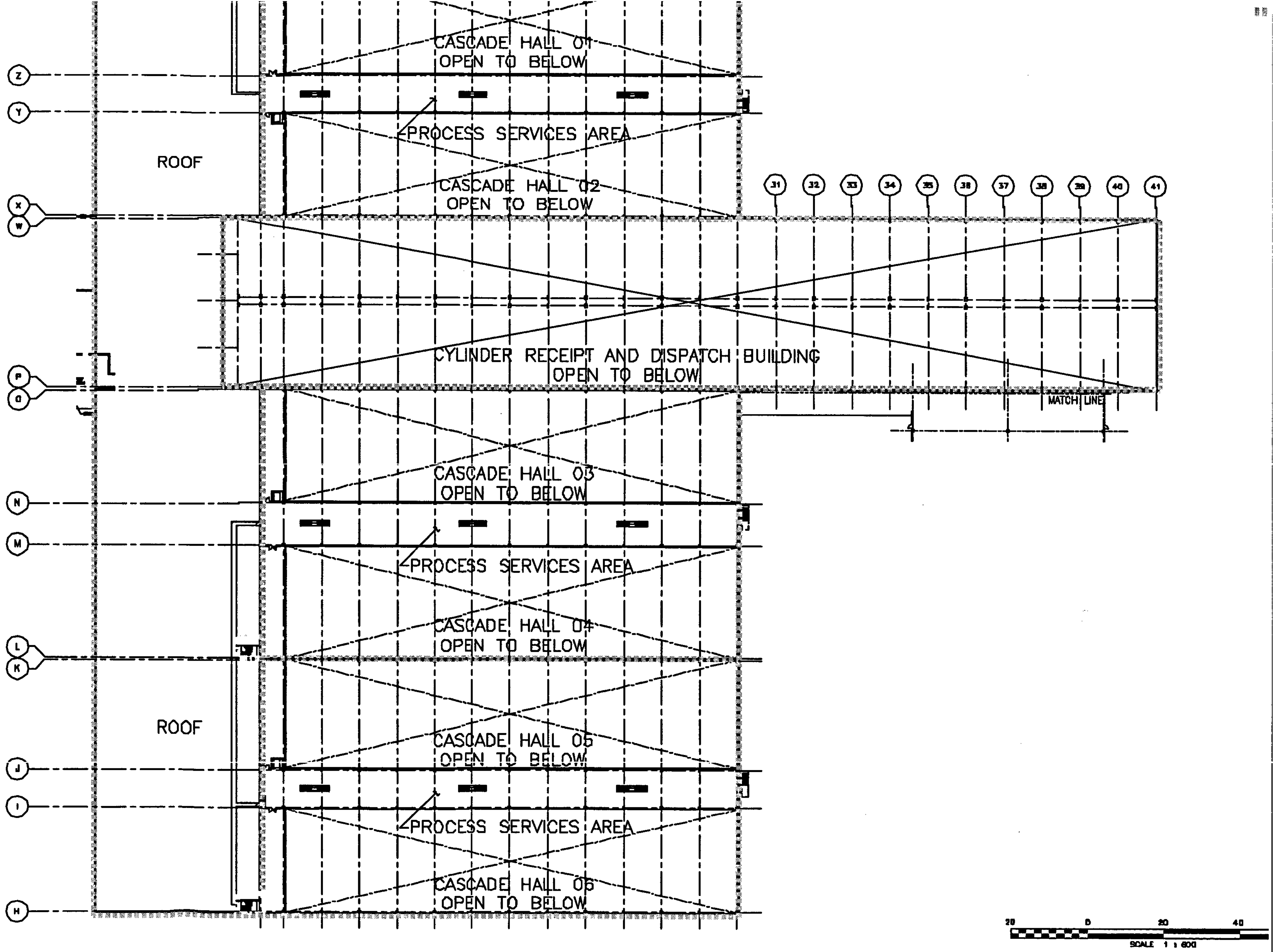

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PROCESS SERVICES AREA
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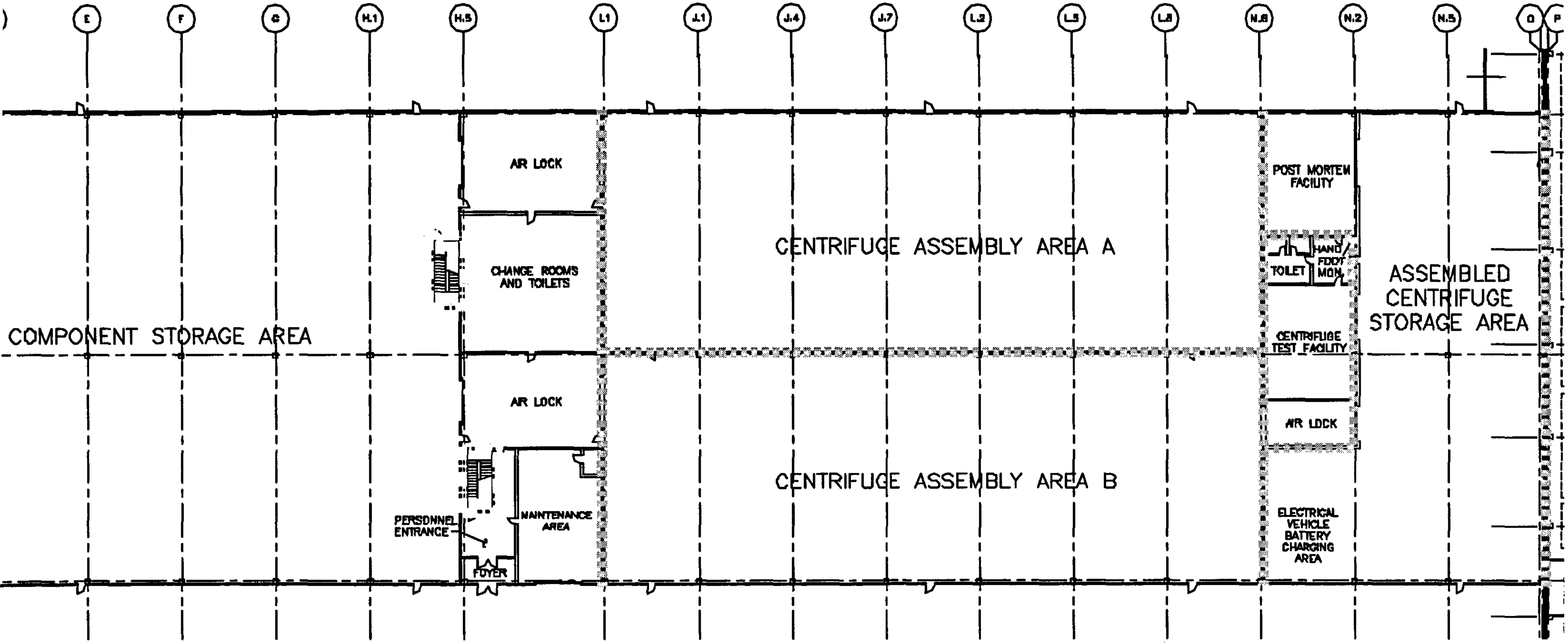

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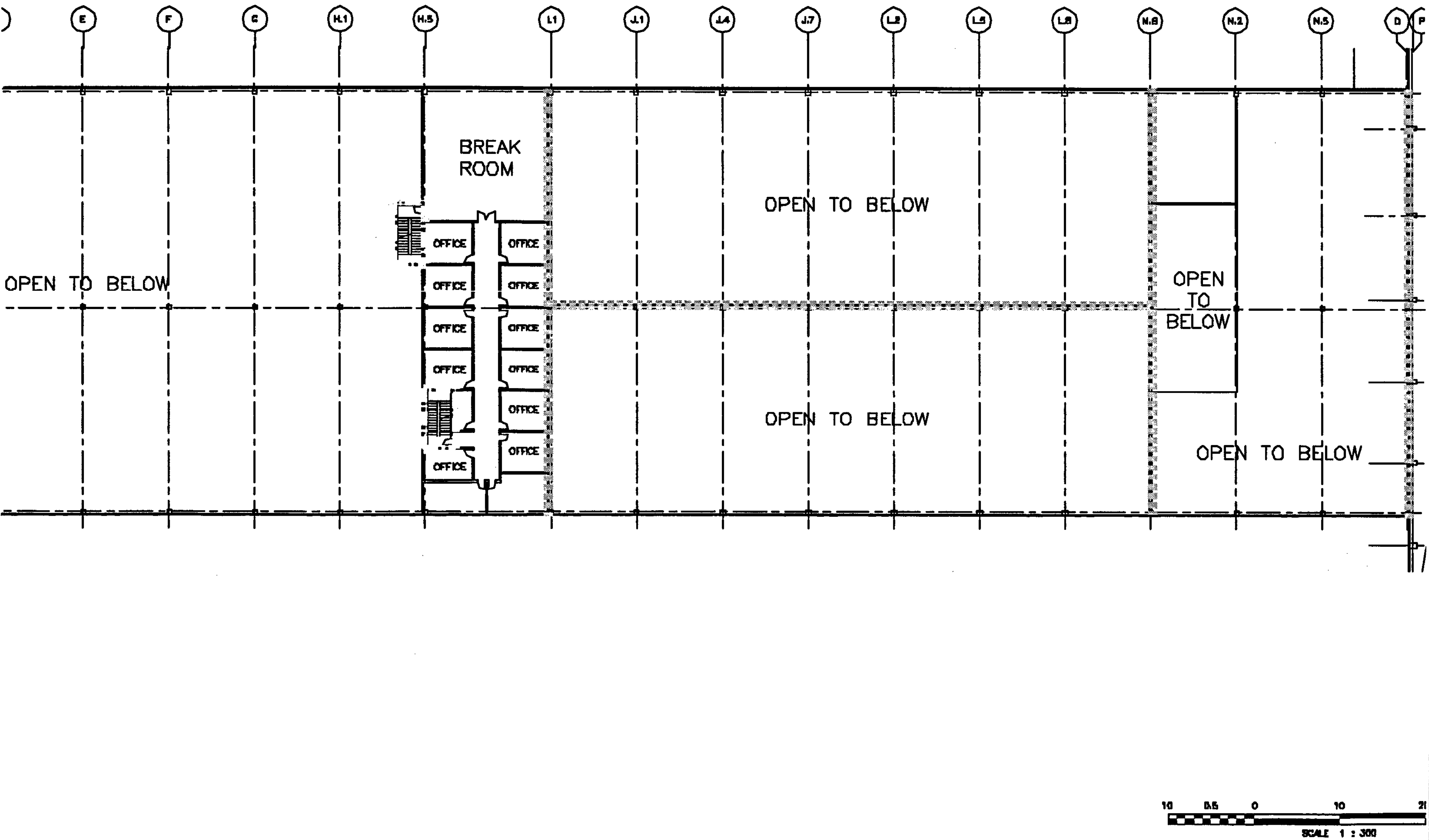
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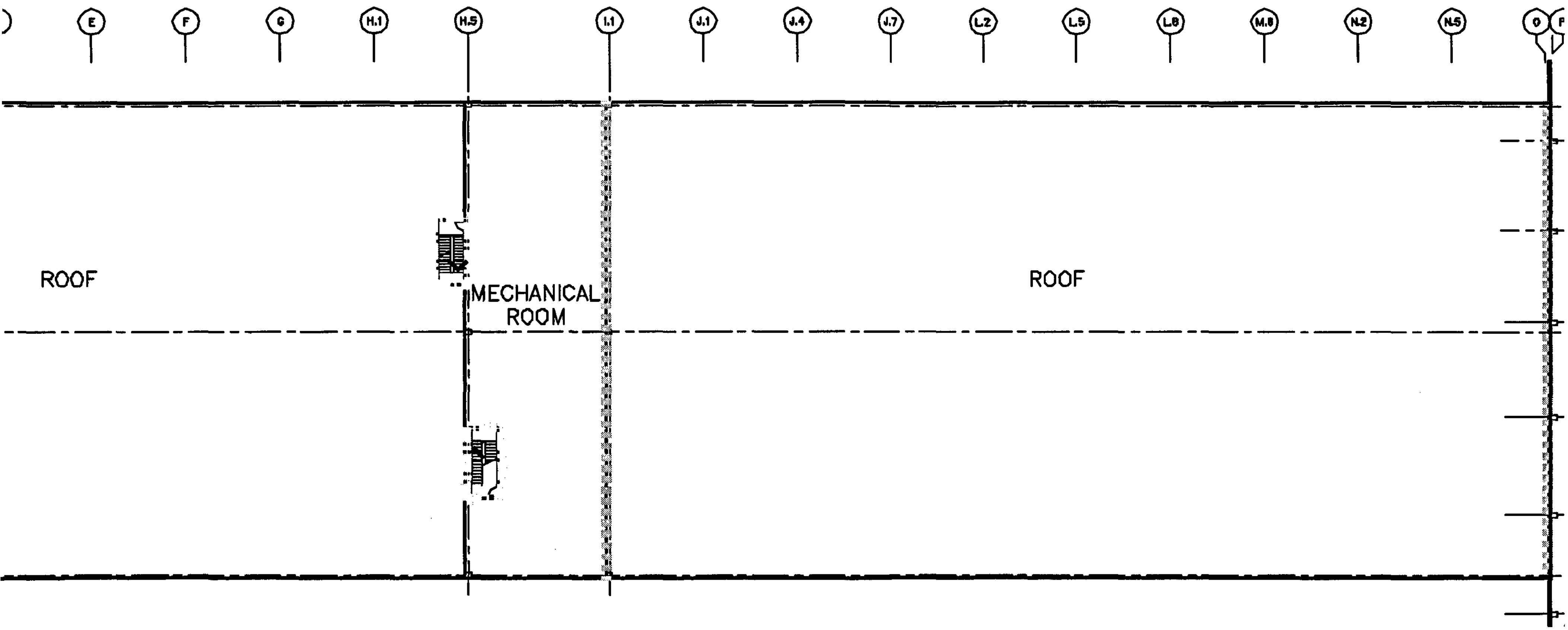
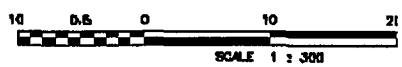
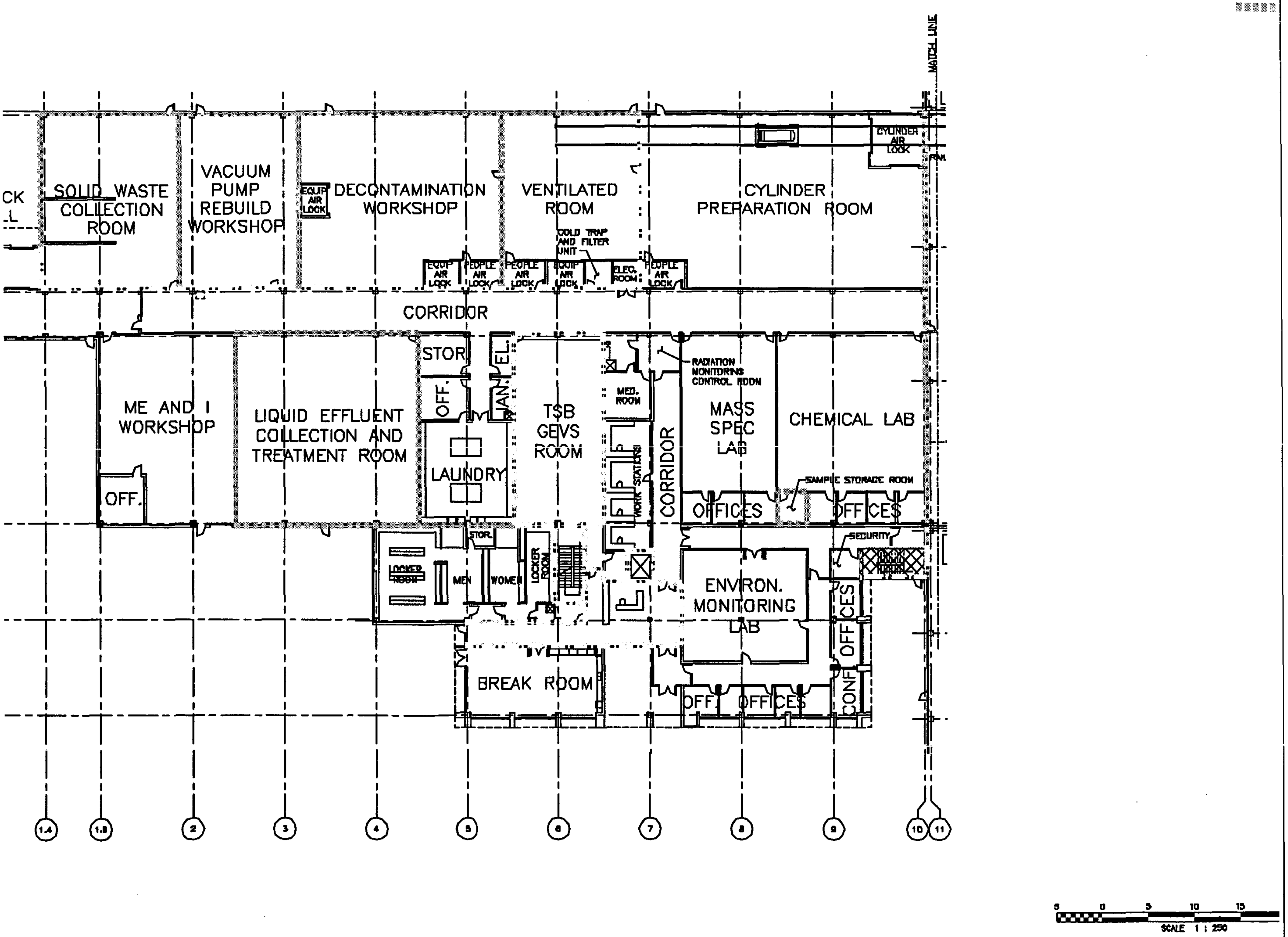


ROTATED CLOCKWISE 90 DEGREES FOR CLARITY




MATCH LINE
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VACUUM PUMP REBUILD WORKSHOP
EQUIP AIR LOCK
DECONTAMINATION WORKSHOP
VENTILATED ROOM
CYLINDER PREPARATION ROOM
CYLINDER AIR LOCK
COLD TRAP AND FILTER UNIT
EQUIP AIR LOCK
PEOPLE AIR LOCK
PEOPLE AIR LOCK
EQUIP AIR LOCK
ELEC. ROOM
PEOPLE AIR LOCK
CORRIDOR
STOR
OFF.
JAN. EL.
ME AND I WORKSHOP
LIQUID EFFLUENT COLLECTION AND TREATMENT ROOM
LAUNDRY
TSB GEVS ROOM
MED. ROOM
RADIATION MONITORING CONTROL ROOM
MASS SPEC LAB
CHEMICAL LAB
CORRIDOR
WORK STATIONS
SAMPLE STORAGE ROOM
OFF.
OFFICES
OFFICES
STOR.
LOCKER ROOM
MEN
WOMEN
LOCKER ROOM
SECURITY
ENVIRON. MONITORING LAB
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CONF.
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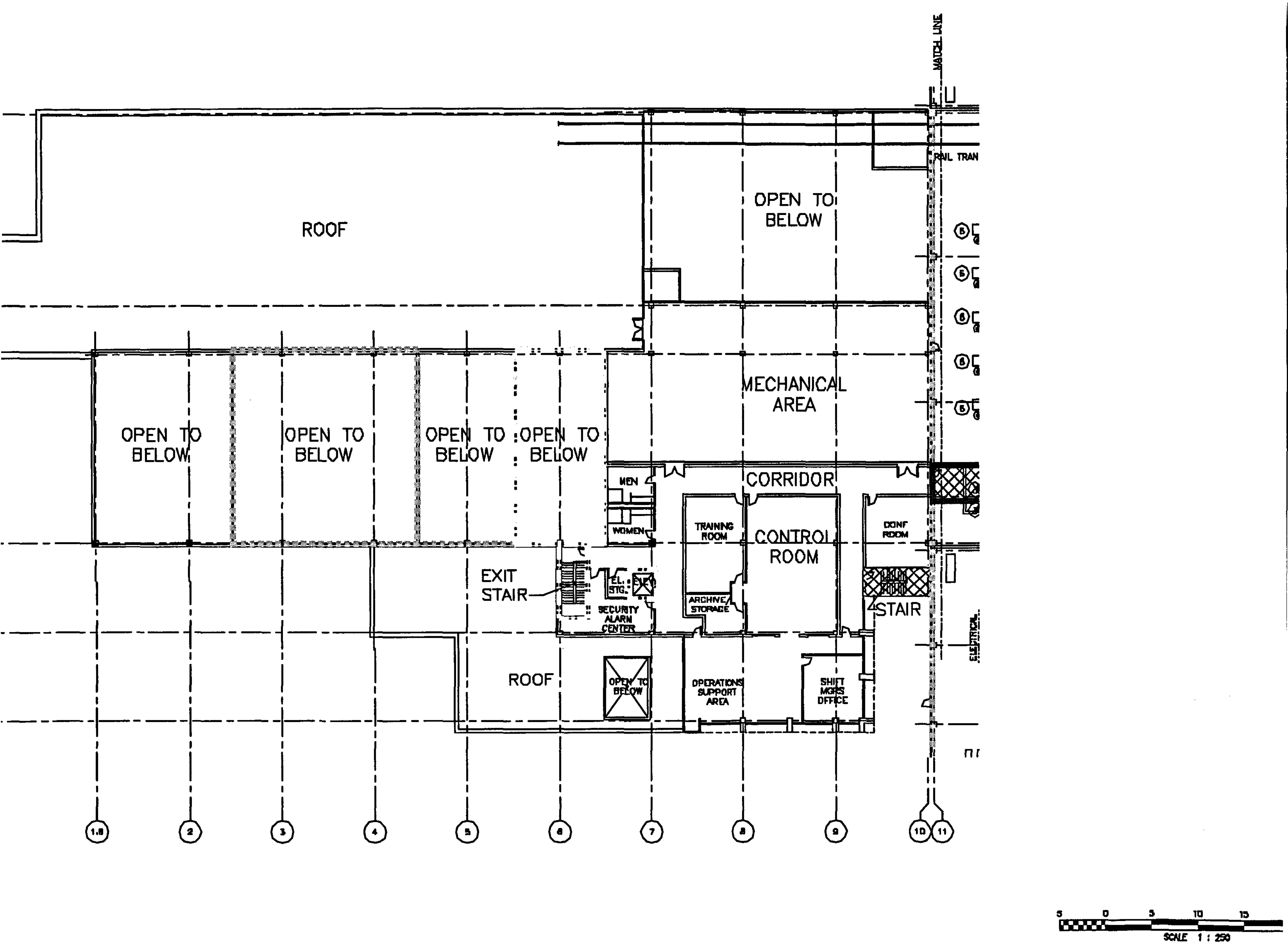

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MEN
WOMEN
TRAINING ROOM
CONTROL ROOM
CONF ROOM
EXIT STAIR
EL. STG.
SECURITY ALARM CENTER
ARCHIVE STORAGE
STAIR
ELECTRICAL
ROOF
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SHIFT MGRS OFFICE
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3. EACH HYDRANT SHALL BE EQUIPPED WITH
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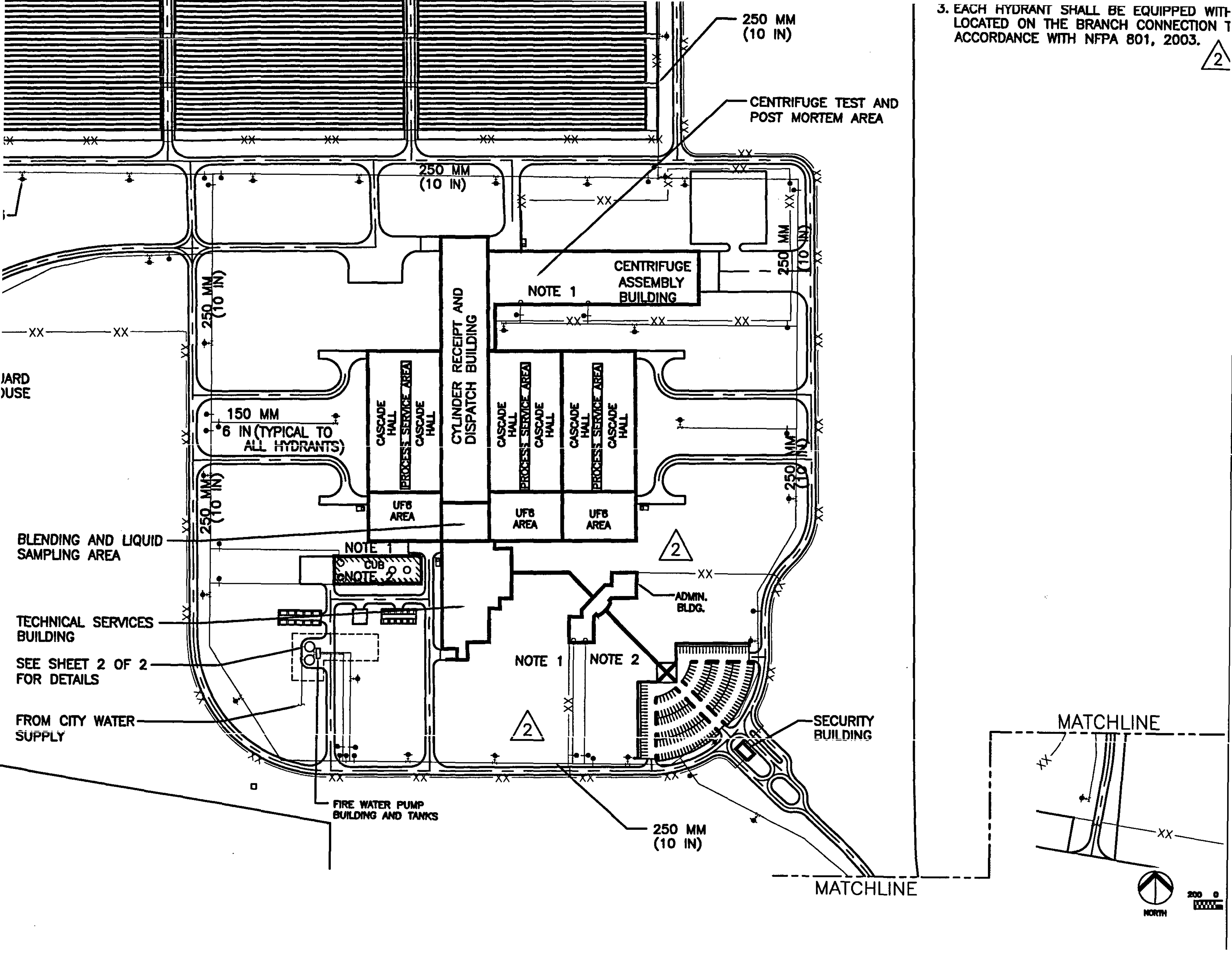
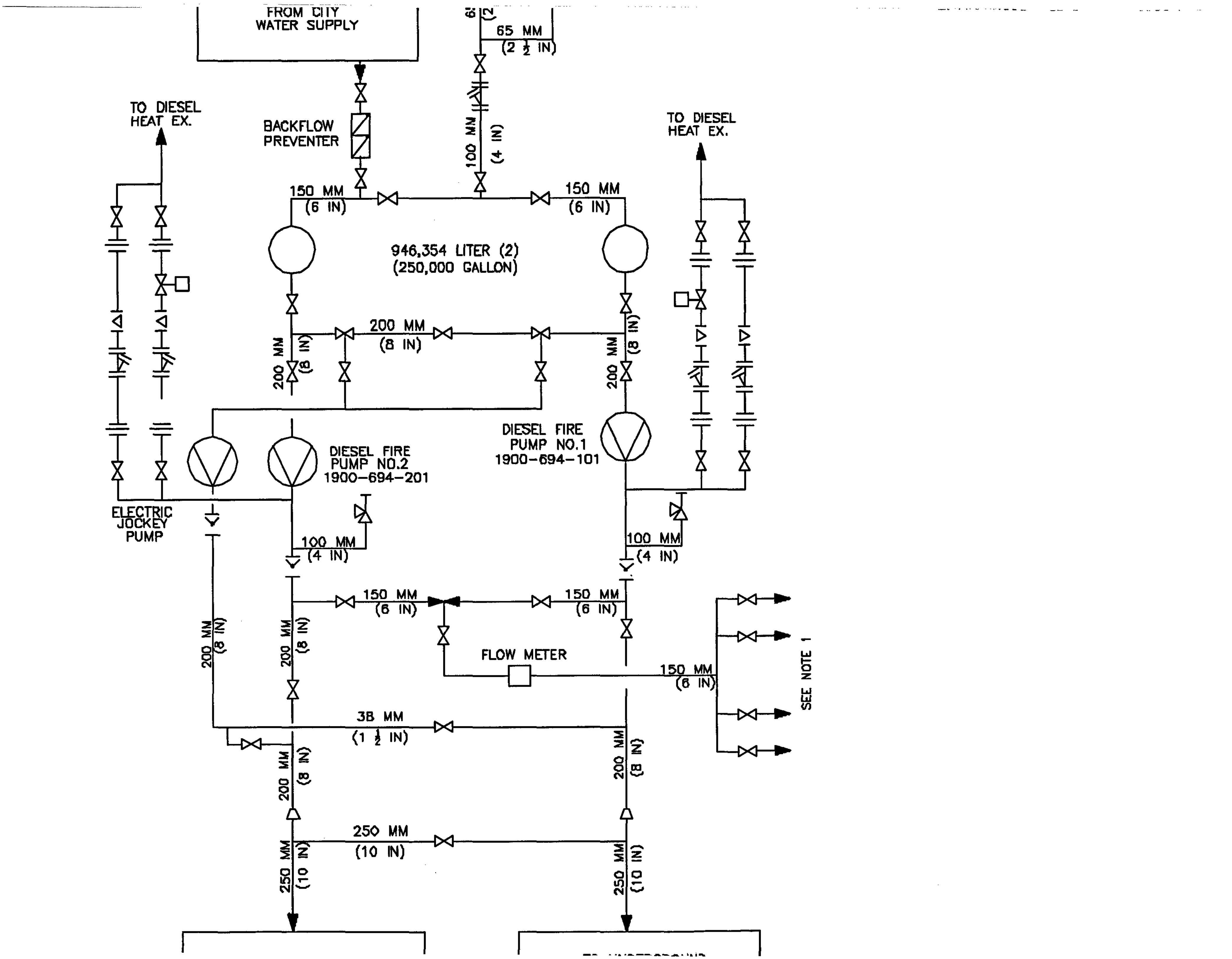
ACCORDANCE WITH NFPA 801, 2003.





FROM CITY
WATER SUPPLY
65 MM
(2 ½ IN)
100 MM
(4 IN)
BACKFLOW
PREVENTER
TO DIESEL
HEAT EX.
TO DIESEL
HEAT EX.
150 MM
(6 IN)
150 MM
(6 IN)
946,354 LITER (2)
(250,000 GALLON)
200 MM
(8 IN)
200 MM
(8 IN)
200 MM
(8 IN)
DIESEL FIRE
PUMP NO.1
1900-694-101
DIESEL FIRE
PUMP NO.2
1900-694-201
ELECTRIC
JOCKEY
PUMP
100 MM
(4 IN)
100 MM
(4 IN)
150 MM
(6 IN)
150 MM
(6 IN)
200 MM
(8 IN)
200 MM
(8 IN)
FLOW METER
150 MM
(6 IN)
SEE NOTE 1
38 MM
(1 ½ IN)
200 MM
(8 IN)
200 MM
(8 IN)
250 MM
(10 IN)
250 MM
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250 MM
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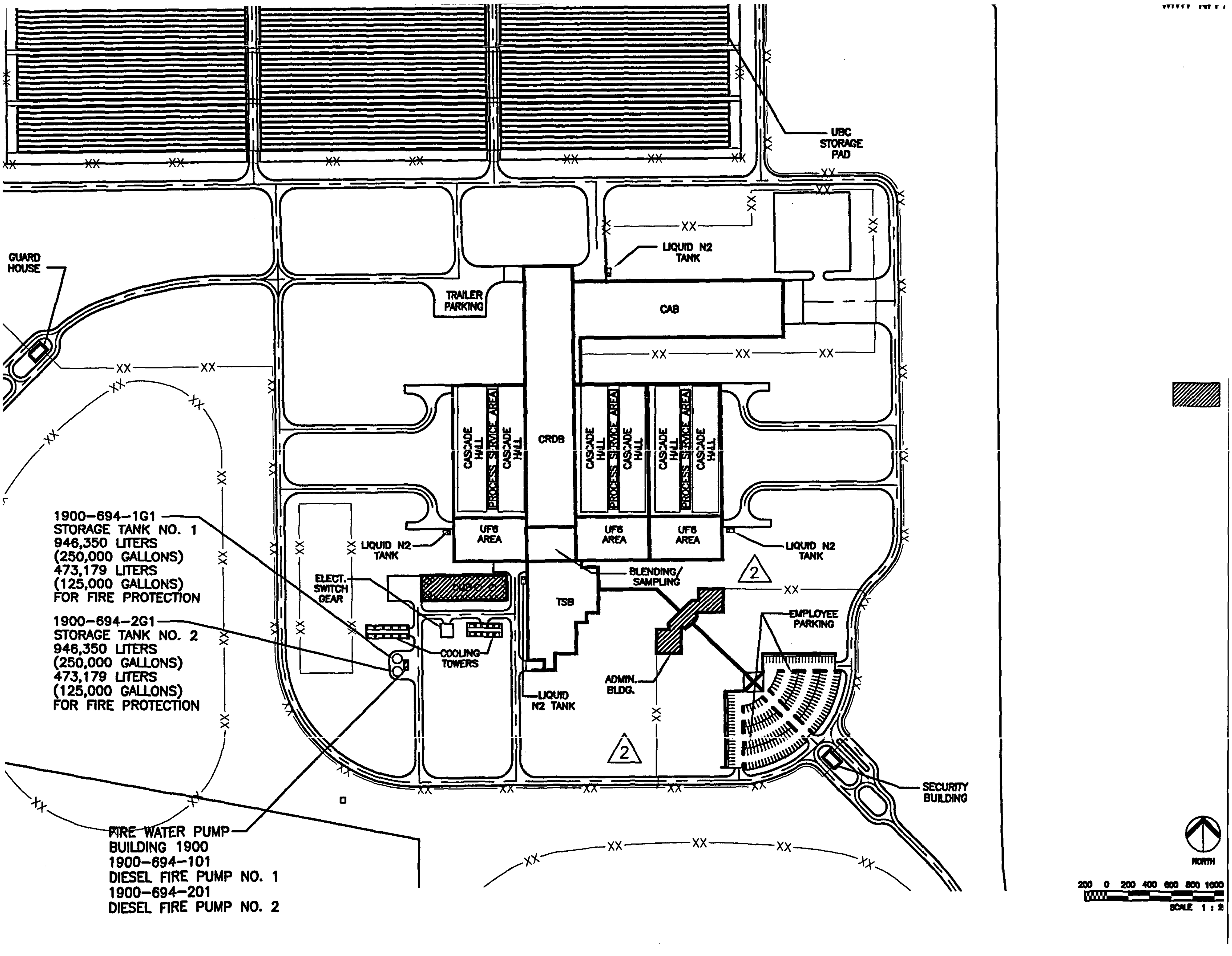
UBC STORAGE PAD
GUARD HOUSE
LIQUID N2 TANK
TRAILER PARKING
CAB
CRDB
CASCADE HALL
PROCESS SERVICE AREA
CASCADE HALL
CASCADE HALL
PROCESS SERVICE AREA
CASCADE HALL
CASCADE HALL
PROCESS SERVICE AREA
CASCADE HALL
UF6 AREA
UF6 AREA
UF6 AREA
LIQUID N2 TANK
LIQUID N2 TANK
BLENDING/ SAMPLING
TSB
ELECT. SWITCH GEAR
COOLING TOWERS
ADMIN. BLDG.
EMPLOYEE PARKING
LIQUID N2 TANK
SECURITY BUILDING
1900-694-1G1
STORAGE TANK NO. 1
946,350 LITERS
(250,000 GALLONS)
473,179 LITERS
(125,000 GALLONS)
FOR FIRE PROTECTION
1900-694-2G1
STORAGE TANK NO. 2
946,350 LITERS
(250,000 GALLONS)
473,179 LITERS
(125,000 GALLONS)
FOR FIRE PROTECTION
FIRE WATER PUMP
BUILDING 1900
1900-694-101
DIESEL FIRE PUMP NO. 1
1900-694-201
DIESEL FIRE PUMP NO. 2
NORTH
200 0 200 400 600 800 1000
SCALE 1 : 2

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8.0 EMERGENCY MANAGEMENT

The plans for coping with emergencies at the National Enrichment Facility are presented in the facility Emergency Plan. The Emergency Plan has been developed in accordance with 10 CFR 70.22(i) (CFR, 2003a) and 10 CFR 40.31(j) (CFR, 2003b). The Emergency Plan conforms to the guidance presented in Regulatory Guide 3.67, Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities, (NRC, 1992). The facility Emergency Plan also addresses the specific acceptance criteria in NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, (NRC, 2002), Chapter 8, Emergency Management.

The Emergency Plan identifies the offsite organizations that reviewed the Emergency Plan pursuant to the requirement in 10 CFR 70.22(i)(4) (CFR, 2003a) and 10 CFR 40.31(j)(4) (CFR, 2003b). Memorandums of Understanding with the off-site organizations are provided in the Emergency Plan.

8.1 REFERENCES

CFR, 2003a. Title 10, Code of Federal Regulations, Section 70.22, Contents of applications, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Section 40.31, Application for specific licenses, 2003.

NRC, 1992. Standard Format and Content of Emergency Plans for Fuel Cycle and Materials Facilities, Regulatory Guide 3.67, U.S. Nuclear Regulatory Commission, January 1992.

NRC, 2002. Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, NUREG-1520, U.S. Nuclear Regulatory Commission, March 2002.

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9.0 ENVIRONMENTAL PROTECTION

Louisiana Energy Services (LES) has prepared documents to demonstrate that its proposed environmental protective measures are adequate to protect the environment and the health and safety of the public as well as comply with the regulatory requirements imposed in 10 CFR 20 (CFR, 2003a), 10 CFR 30 (CFR, 2003b), 10 CFR 40 (CFR, 2003c), 10 CFR 51 (CFR, 2003d), and 10 CFR 70 (CFR, 2003e). The Environmental Report (ER) from LES' previous application (LES, 1994) was reviewed and information that was unchanged and found acceptable by the Nuclear Regulatory Commission (NRC) in NUREG-1484 (NRC, 1994) has been noted in the present ER.

Summarized below are the chapter section, general information category, the corresponding regulatory requirement, and the NUREG-1520 (NRC, 2002) section identifying the NRC acceptance criteria.

Chapter Section	Information Category	10 CFR Citation	NUREG-1520 Reference
9.1	**Environmental Report**	70.21(h)	9.4.3.1.1
9.1.1	Date of Application	70.21(f)	9.4.3.1.1(1)
9.1.2	Environmental Considerations	51.45(b)	9.4.3.1.1(2)
9.1.3	Analysis of Effects of Proposed Action and Alternatives	51.45(c)	9.4.3.1.1(3)
9.1.4	Status of Compliance	51.45(d)	9.4.3.1.1(4)
9.1.5	Adverse Information	51.45(e)	9.4.3.1.1(5)
9.2	**Environmental Protection Measures**	70.22(a)(8)	9.4.3.2
9.2.1	Radiation Safety	20.1101(a)	9.4.3.2.1
	• ALARA Controls and Reports	20.1101(d)	9.4.3.2.1(1)-(3)
	• Waste Minimization	20.1406	9.4.3.2.1(4)
9.2.2	Effluent and Environmental Controls and Monitoring	70.59(a)(1)	9.4.3.2.2
9.2.2.1	Effluent Monitoring	20.1501(a)	9.4.3.2.2(1)
9.2.2.2	Environmental Monitoring	20.1501(a)	9.4.3.2.2(2)
9.2.2.3	ISA Summary	70.65(b)	9.4.3.2.2(3)

This Safety Analysis Report (SAR) Chapter documents the potential environmental impacts associated with construction and operation of the NEF and indicates that adverse impacts are small. These impacts are outweighed by the substantial socioeconomic benefits associated with plant construction and operation. Additionally, the NEF will meet the underlying need for additional reliable and economical uranium enrichment capacity in the United States, thereby serving important energy and national security policy objectives. Accordingly, because the impacts of the proposed NEF are minimal and acceptable, and the benefits are desirable, the no-action alternative may be rejected in favor of the proposed action.

9.1 ENVIRONMENTAL REPORT

LES has prepared an Environmental Report (ER) that meets the requirements contained in 10 CFR Part 51 (CFR, 2003d), Subpart A. In particular, the ER addresses the requirements in 10 CFR 51.45(b)-(e) (CFR, 2003f) and follows the general format of NUREG-1748 (NRC, 2003).

The ER presents the proposed action, purpose of the proposed action, and applicable regulatory requirements (Chapter 1), discusses alternatives (Chapter 2), describes the facility and the affected environment (Chapter 3), and potential impacts of the proposed action (Chapter 4). Mitigation measures are described in Chapter 5, environmental measurements and monitoring programs in Chapter 6, a cost-benefit analysis in Chapter 7, and a summary of environmental consequences in Chapter 8. References and preparers are listed in Chapters 9 and 10, respectively.

9.1.1 Date of Application

The effective date of the ER is December 16, 2003. As required by 10 CFR 70.21(f) (CFR, 2003g), this date is at least nine months before facility construction is scheduled to begin in 2006.

9.1.2 Environmental Considerations

Applicant's ER adequately addresses the requirements of 10 CFR 51.45(b) (CFR, 2003f) as follows:

9.1.2.1 Description of the Proposed Action

The proposed action, described in ER Section 1.1, Proposed Action, is the issuance of an NRC specific license under 10 CFR 30 (CFR, 2003b), 10 CFR 40 (CFR, 2003c) and 10 CFR 70 (CFR, 2003e) to possess and use byproduct material, source material and special nuclear material (SNM) and to construct and operate a uranium enrichment facility in Lea County, New Mexico. The enriched uranium is intended for use primarily in domestic commercial nuclear power plants.

Significant characteristics of the facility are described in ER Chapters 1, Introduction of the Environmental Report and Chapter 3, Description of Affected Environment. Major site features, along with plant design and operating parameters are included. A discussion of how the special nuclear material (SNM), in this case uranium hexafluoride (UF_6), will be processed to produce enriched uranium-235 (^{235}U) is described in ER Section 1.2, Proposed Action, which also includes the proposed project schedule.

9.1.2.2 Purpose of Proposed Action

ER Section 1.2, Purpose and Need for the Proposed Action, demonstrates the need for the facility. The demonstration provides the

- Quantities of SNM used for domestic benefit

- A projection of domestic and foreign requirements for services
- Alternative sources of supply for LES' proposed services.

ER Section 1.2, Purpose and Need for the Proposed Action, also discusses if delay of the facility occurs, the effects to the nation's energy program or LES's business such as loss of contracts.

9.1.2.3 Description of the Affected Environment

Chapter 3 of the ER contains detailed descriptions of the affected environment. The chapter provides a baseline characterization of the site and its environs prior to any disturbances associated with construction or operation of the facility. The following topics and corresponding ER chapter section include:

- Site location (including longitude and latitude) and facility layout (1.2)
- Regional demography (3.10) and land use (3.1)
- Socioeconomic information (3.10), including low-income and minority populations within 130 km^2 (50 mi^2) as directed by NUREG-1748 (4.11)
- Regional historic (3.8), archeological (3.8), architectural (3.9), scenic (3.9), cultural (3.8), and natural landmarks (3.9)
- Local meteorology and air quality (3.6)
- Local surface water and ground water hydrology (3.4)
- Regional geology and seismology (3.3)
- Local terrestrial and aquatic ecology (3.5).

The baseline descriptions presented are from the most current information available. It was gathered from Federal, State, and County sources along with existing on-site data. Therefore, the information represents both seasonal and long-term environmental trends.

9.1.2.4 Discussion of Considerations

Three ER chapters discuss the potential environmental impacts relating to the proposed action. Chapter 4 details environmental and socioeconomic effects due to site preparation and facility construction and operation. Chapter 2 describes alternatives to the proposed action, including siting and designs. Chapter 7 provides a discussion of the costs and benefits for each alternative as well as the relationship between short-term use and long-term productivity of the environment, and resources committed. In addition, Chapter 8 provides a summary of environmental consequences from all actions. The associated regulatory criteria and corresponding ER section are as follows.

A. Impact of the Proposed Action on the Environment

- Effects of site preparation and construction on land (4.1) and water use (4.4)

- Effects of facility operation on human population (including consideration of occupation and public radiation exposure) and important biota (4.10, 4.11, and 4.12)

- Any irreversible commitments of resources because of site preparation and facility construction and operation, such as destruction of wildlife habitat, removal of land from agriculture, and diversion of electrical power (4.1, 7.0, and 8.2)
- Plans and policies regarding decommissioning and dismantling at the end of the facility's life (8.9)
- Environmental effects of the transportation of radioactive materials to and from the site (4.2)
- Environmental effects of accidents (4.12)
- Impacts on air (4.6) and water quality (4.4)
- Impacts on cultural and historic resources (4.8).

B. Adverse Environmental Effects

Three chapters in the ER discuss adverse environmental effects. Refer to Section 9.1.5 below for additional detail on the associated ER chapters and topics.

C. Alternatives to the Proposed Action

ER Chapter 2 provides a complete description of alternatives to the proposed action. Included are the no action alternative scenarios as well as the siting criteria and technical design requirements in sufficient detail to allow a fair and reasonable comparison between the alternatives.

D. Relationship between Short- and Long-term Productivity

ER Chapter 7, the cost-benefit analysis, included the consideration of the short-term uses and productivity of the site during the active life of the facility. No adverse impacts on the long-term productivity of the environment after decommissioning of the facility have been identified. The European experience at the Almelo enrichment plant demonstrates that a centrifuge technology site can be returned to a greenfield site for use without restriction.

E. Irreversible and Irretrievable Commitments of Resources

Irreversible environmental commitments and irretrievable material resources also are included in the cost-benefit analysis in ER Chapter 7. They are part of the capital costs associated with the land and facility and operating and maintenance costs. No significant commitments are involved with the proposed action. The site should be available for unrestricted use following decommissioning. Some components may be reused or sold as scrap during the plant life or following decommissioning.

9.1.3 Analysis of Effects of Proposed Action and Alternatives

ER Chapter 2 discusses the analysis of effects of the proposed action and alternatives in accordance with 10 CFR 51.45(c) (CFR, 2003f). The analysis considers and balances the environmental effects of the proposed action and alternatives available to reduce or avoid both environmental and socioeconomic effects and other benefits of the proposed action.

9.1.4 Status of Compliance

ER Section 1.3 summarizes, as required in 10 CFR 51.45(d) (CFR, 2003f), the applicability of environmental regulatory requirements, permits, licenses, or approvals as well as the current status of each on the effective date of the ER.

Many federal laws and regulations apply to the facility during site assessment, construction, and operation. Some of these laws require permits from, consultations with, or approvals by, other governing or regulatory agencies. Some apply only during certain phases of facility development, rather than the entire life of the facility. Federal statutes and regulations (non-nuclear) have been reviewed to determine their applicability to the facility site assessment, construction, and operation.

9.1.5 Adverse Information

In accordance with 10 CFR 51.45(e) (CFR, 2003f), various sections throughout the ER discuss adverse environmental effects. In particular, Chapter 4 details environmental and socioeconomic effects due to site preparation and facility construction and operation. Chapter 2 compares potential impacts from alternatives. Lastly, Chapter 8 provides a summary of environmental consequences from all actions.

9.2 ENVIRONMENTAL PROTECTION MEASURES

LES is committed to protecting the public, plant workers, and the environment from the harmful effects of ionizing radiation due to plant operation. Accordingly, LES is firmly committed to the "As Low As Reasonably Achievable," (ALARA) philosophy for all operations involving source, byproduct, and special nuclear material. This commitment is reflected in written procedures and instructions for operations involving potential exposures of personnel to radiation (both internal and external hazards) and the facility design. Written procedures for effluent monitoring address the need for periodic (monthly) dose assessment projections to members of the public to ensure that potential radiation exposures are kept ALARA (i.e., not in excess of 0.1 mSv/yr (10 mrem/yr)) in accordance with 10 CFR 20.1101(d).

Part of LES's environmental protective measures are described in the ER. In particular, Chapter 4 discusses the anticipated results of the radiation protection program with regard to ALARA goals and waste minimization. Chapter 6 discusses the environmental controls and monitoring program.

A detailed description of LES' radiation protection program is included separately in this License Application as Safety Analysis Report (SAR) Chapter 4. Similarly, LES's provisions for a qualified and trained staff, which also is part of the environmental protection measures required, are described separately in the SAR as part of Chapter 11.

9.2.1 Radiation Safety

The four acceptance criteria that describe the facility radiation safety program are divided between two License Application documents. SAR Chapter 4 describes:

- Radiological (ALARA) Goals for Effluent Control
- ALARA Reviews and Reports to Management.

ER Chapter 4, Environmental Impacts, addresses:

- Effluents controls to maintain public doses ALARA, and
- Waste Minimization.

In particular, ER Section 4.12 describes public and occupational health effects from both non-radiological and radiological sources. This section specifically addresses calculated total effective dose equivalent to an average member of critical groups or calculated average annual concentration of radioactive material in gaseous and liquid effluent to maintain compliance with 10 CFR 20 (CFR, 2003a).

ER Section 4.13 contains a discussion on facility waste minimization that identifies process features and systems to reduce or eliminate waste. It also describes methods to minimize the volume of waste.

9.2.2 Effluent and Environmental Controls and Monitoring

LES has designed an environmental monitoring program to provide comprehensive data to monitor the facility's impact on the environment. The preoperational program will focus on collecting data to establish baseline information useful in evaluating changes in potential environmental conditions caused by facility operation. The preoperational program will be initiated at least two years prior to facility operation.

The operational program will monitor to ensure facility emissions are maintained ALARA. Monitoring will be of appropriate pathways up to a 2-mile radius beyond the site boundary.

ER Chapter 6 describes environmental measurement and monitoring programs as they apply to preoperation (baseline), operation, and decommissioning conditions for both the proposed action and each alternative.

9.2.2.1 Effluent Monitoring

ER Section 6.1 presents information relating to the facility radiological monitoring program. This section describes the location and characteristics of radiation sources and radioactive effluent (liquid and gaseous). It also describes the various elements of the monitoring program, including:

- Number and location of sample collection points
- Measuring devices used
- Pathway sampled or measured
- Sample size, collection frequency and duration
- Method and frequency of analysis, including lower limits of detection.

Based on recorded plant effluent data, dose projections to members of the public will be performed monthly to ensure that the annual dose to members of the public does not exceed the ALARA constraint of 0.1 mSv/yr (10 mrem/yr). If the monthly dose impact assessment indicates a trend in effluent releases that, if not corrected, could cause the ALARA constraint to be exceeded, appropriate corrective action will be initiated to reduce the discharges to assure that subsequent releases will be in compliance with the annual dose constraint. In addition, an evaluation of the need for increased sampling will be performed. Corrective actions may include, for example, change out of Separation Building or Technical Services Building Gaseous Effluent Vent System filters, replacement of spent cleanup resins for liquid waste or reprocessing collected waste prior to release to the Treated Effluent Evaporative Basin.

Lastly, this section justifies the choice of sample locations, analyses, frequencies, durations, sizes, and lower limits of detection.

9.2.2.2 Environmental Monitoring

ER Section 6.1 also includes information relating to the facility environmental monitoring program. The information presented is the same as that included in the effluent monitoring program, i.e., number and location of sample collection points, etc.

9.2.3 Integrated Safety Analysis

LES has prepared an integrated safety analysis (ISA) in accordance with 10 CFR 70.60 (CFR, 2003h). The ISA

- Provides a complete list of the accident sequences that if uncontrolled could result in radiological and non-radiological releases to the environment with intermediate or high consequences
- Provides reasonable estimates for the likelihood and consequences of each accident identified
- Applies acceptable methods to estimate environmental effects that may result from accidental releases.

The ISA also

- Identifies adequate engineering and/or administrative controls for each accident sequence of environmental significance
- Assures adequate levels are afforded so those items relied on for safety (IROFS) will satisfactorily perform their safety functions.

The ISA demonstrates that the facility and its operations have adequate engineering and/or administrative controls in place to prevent or mitigate high and intermediate consequences from the accident sequences identified and analyzed.

9.3 REFERENCES

CFR, 2003a. Title 10, Code of Federal Regulations, Part 20, Standards for Protection Against Radiation, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Part 30, Rules of General Applicability to Domestic Licensing of Byproduct Material, 2003.

CFR, 2003c. Title 10, Code of Federal Regulations, Part 40, Domestic Licensing of Source Material, 2003.

CFR, 2003d. Title 10, Code of Federal Regulations, Part 51, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions, 2003.

CFR, 2003e. Title 10, Code of Federal Regulations, Part 70, Domestic Licensing of Special Nuclear Material, 2003.

CFR, 2003f. Title 10, Code of Federal Regulations, Section 51.45, Environmental report, 2003.

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10.0 DECOMMISSIONING

This chapter presents the National Enrichment Facility (NEF) Decommissioning Funding Plan. The Decommissioning Funding Plan has been developed following the guidance provided in NUREG-1757 (NRC, 2003). This Decommissioning Funding Plan is similar to the decommissioning funding plan for the Claiborne Enrichment Center (CEC) approved by the NRC in NUREG-1491 (NRC, 1994).

Louisiana Energy Services (LES) commits to decontaminate and decommission the enrichment facility and the site at the end of its operation so that the facility and grounds can be released for unrestricted use. The Decommissioning Funding Plan will be reviewed and updated as necessary at least once every three years starting from the time of issuance of the license. Prior to facility decommissioning, a Decommissioning Plan will be prepared in accordance with 10 CFR 70.38 (CFR, 2003a) and submitted to the NRC for approval.

This chapter fulfills the applicable provisions of NUREG-1757 (NRC, 2003) through submittal of information in tabular form as suggested by the NUREG. Therefore a matrix showing compliance requirements and commitments is not provided herein.

10.1 SITE-SPECIFIC COST ESTIMATE

10.1.1 Cost Estimate Structure

The decommissioning cost estimate is comprised of three basic parts that include:

- A facility description
- The estimated costs (including labor costs, non-labor costs, and a contingency factor)
- Key assumptions.

10.1.2 Facility Description

The NEF is fully described in other sections of this License Application and the NEF Integrated Safety Analysis Summary. Information relating to the following topics can be found in the referenced chapters listed below:

A general description of the facility and plant processes is presented in Chapter 1, General Information. A detailed description of the facility and plant processes is presented in the NEF Integrated Safety Analysis Summary.

A description of the specific quantities and types of licensed materials used at the facility is provided in Chapter 1, Section 1.2, Institutional Information.

A general description of how licensed materials are used at the facility is provided in Chapter 1, General Information.

10.1.3 Decommissioning Cost Estimate

10.1.3.1 Summary of Costs

The decommissioning cost estimate for the NEF is approximately $837 million (January, 2002 dollars). The decommissioning cost estimate and supporting information are presented in Tables 10.1-1A through 10.1-14, consistent with the applicable provisions of NUREG-1757, NMSS Decommissioning Standard Review Plan (NRC, 2003).

More than 97% of the decommissioning costs (except tails disposition costs) for the NEF are attributed to the dismantling, decontamination, processing, and disposal of centrifuges and other equipment in the Separations Building Modules, which are considered classified. Given the classified nature of these buildings, the data presented in the Tables at the end of this chapter has been structured to meet the applicable NUREG-1757 (NRC, 2003) recommendations, to the extent practicable. However, specific information such as numbers of components and unit rates have been intentionally excluded to protect the classified nature of the data.

The remaining 3% of the decommissioning costs are for the remaining systems and components in other buildings. Since these costs are small in relation to the overall cost estimate, the cost data for these systems has also been summarized at the same level of detail as that for the Separations Building Modules.

The decommissioning project schedule is presented in Figure 10.1-1, National Enrichment Facility – Conceptual Decommissioning Schedule. Dismantling and decontamination of the equipment in the three Separations Building Modules will be conducted sequentially (in three phases) over a nine year time frame. Separations Building Module 1 will be decommissioned during the first three-year period, followed by Separations Building Module 2, and then Separations Building Module 3. Termination of Separations Module 3 operations will mark the end of uranium enrichment operations at the NEF. Decommissioning of the remaining plant systems and buildings will begin after Separations Building Module 3 operations have been permanently terminated.

10.1.3.2 Major Assumptions

Key assumptions underlying the decommissioning cost estimate are listed below:

- Inventories of materials and wastes at the time of decommissioning will be in amounts that are consistent with routine plant operating conditions over time.
- Costs are not included for the removal or disposal of non-radioactive structures and materials beyond that necessary to terminate the NRC license.
- Credit is not taken for any salvage value that might be realized from the sale of potential assets (e.g., recovered materials or decontaminated equipment) during or after decommissioning.
- Decommissioning activities will be performed in accordance with current day regulatory requirements.
- LES will be the Decommissioning Operations Contractor (DOC) for all decommissioning operations. However, in the event that LES is not able to fulfill this role, an adjustment to account for use of a third party for performing decommissioning operations is provided in Table 10.1-14, Total Decommissioning Costs.
- Decommissioning costs, with the exception of tails disposition costs, are presented in January 2002 dollars. In Table 10.1-14, tails disposition costs are presented in January 2004 dollars. In addition, the costs of decommissioning presented in Table 10.1-14 are escalated from January 2002 dollars to January 2004 dollars to provide the total decommissioning costs in January 2004 dollars.

10.1.4 Decommissioning Strategy

The plan for decommissioning is to promptly decontaminate or remove all materials from the site which prevent release of the facility for unrestricted use. This approach, referred to in the industry as DECON (i.e., immediate dismantlement), avoids long-term storage and monitoring of wastes on site. The type and volume of wastes produced at the NEF do not warrant delays in waste removal normally associated with the SAFSTOR (i.e., deferred dismantlement) option.

At the end of useful plant life, the enrichment facility will be decommissioned such that the site and remaining facilities may be released for unrestricted use as defined in 10 CFR 20.1402 (CFR, 2003b). Enrichment equipment will be removed; only building shells and the site infrastructure will remain. All remaining facilities will be decontaminated where needed to acceptable levels for unrestricted use. Confidential and Secret Restricted Data material, components, and documents will be destroyed and disposed of in accordance with the facility Standard Practice Procedures Plan for the Protection of Classified Matter.

Depleted UF_6 (tails), if not already sold or otherwise disposed of prior to decommissioning, will be disposed of in accordance with regulatory requirements. Radioactive wastes will be disposed of in licensed low-level radioactive waste disposal sites. Hazardous wastes will be treated or disposed of in licensed hazardous waste facilities. Neither tails conversion (if done), nor disposal of radioactive or hazardous material will occur at the plant site, but at licensed facilities located elsewhere.

Following decommissioning, no part of the facilities or site will remain restricted to any specific type of use.

Activities required for decommissioning have been identified, and decommissioning costs have been estimated. Activities and costs are based on actual decommissioning experience in Europe. Urenco has a fully operational dismantling and decontamination facility at its Almelo, Netherlands plant. Data and experience from this operating facility have allowed a very realistic estimation of decommissioning requirements. Using this cost data as a basis, financial arrangements are made to cover all costs required for returning the site to unrestricted use. Updates on cost and funding will be provided periodically and will include appropriate treatment for any replacement equipment. A detailed Decommissioning Plan will be submitted at a later date in accordance with 10 CFR 70.38 (CFR, 2003a).

The remaining subsections describe decommissioning plans and funding arrangements, and provide details of the decontamination aspects of the program. This information was developed in connection with the decommissioning cost estimate. Specific elements of the planning may change with the submittal of the decommissioning plan required at the time of license termination.

10.1.5 Decommissioning Design Features

10.1.5.1 Overview

Decommissioning planning begins with ensuring design features are incorporated into the plant's initial design that will simplify eventual dismantling and decontamination. The plans are implemented through proper management and health and safety programs. Decommissioning policies address radioactive waste management, physical security, and material control and accounting.

Major features incorporated into the facility design that facilitate decontamination and decommissioning are described below.

10.1.5.2 Radioactive Contamination Control

The following features primarily serve to minimize the spread of radioactive contamination during operation, and therefore simplify eventual plant decommissioning. As a result, worker exposure to radiation and radioactive waste volumes are minimized as well.

- Certain activities during normal operation are expected to result in surface and airborne radioactive contamination. Specially designed rooms are provided for these activities to preclude contamination spread. These rooms are isolated from other areas and are provided with ventilation and filtration. The Solid Waste Collection Room, Ventilated Room and the Decontamination Workshop meet these specific design requirements.

- All areas of the plant are sectioned off into Unrestricted and Restricted Areas. Restricted Areas limit access for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Radiation Areas and Airborne Contamination Areas have additional controls to inform workers of the potential hazard in the area and to help prevent the spread of contamination. All procedures for these areas fall under the Radiation Protection Program, and serve to minimize the spread of contamination and simplify the eventual decommissioning.

- Non-radioactive process equipment and systems are minimized in locations subject to potential contamination. This limits the size of the Restricted Areas and limits the activities occurring inside these areas.

- Local air filtration is provided for areas with potential airborne contamination to preclude its spread. Fume hoods filter contaminated air in these areas.

- Curbing, pits, or other barriers are provided around tanks and components that contain liquid radioactive wastes. These serve to control the spread of contamination in case of a spill.

10.1.5.3 Worker Exposure and Waste Volume Control

The following features primarily serve to minimize worker exposure to radiation and minimize radioactive waste volumes during decontamination activities. As a result, the spread of contamination is minimized as well.

- During construction, a washable epoxy coating is applied to floors and walls that might be radioactively contaminated during operation. The coating will serve to lower waste volumes during decontamination and simplify the decontamination process. The coating is applied to floors and walls that might be radioactively contaminated during operation that are located in the Restricted Areas.

- Sealed, nonporous pipe insulation is used in areas likely to be contaminated. This will reduce waste volume during decommissioning.

- Ample access is provided for efficient equipment dismantling and removal of equipment that may be contaminated. This minimizes the time of worker exposure.
- Tanks are provided with accesses for entry and decontamination. Design provisions are also made to allow complete draining of the wastes contained in the tanks.
- Connections in the process systems provided for required operation and maintenance allow for thorough purging at plant shutdown. This will remove a significant portion of radioactive contamination prior to disassembly.
- Design drawings, produced for all areas of the plant, will simplify the planning and implementing of decontamination procedures. This in turn will shorten the durations that workers are exposed to radiation.
- Worker access to contaminated areas is controlled to assure that workers wear proper protective equipment and limit their time in the areas.

10.1.5.4 Management Organization

An appropriate organizational strategy will be developed to support the phased decommissioning schedule discussed in Section 10.1.3.1, Summary of Costs. The organizational strategy will ensure that adequate numbers of experienced and knowledgeable personnel are available to perform the technical and administrative tasks required to decommission the facility.

LES intends to be the prime Decommissioning Operations Contractor (DOC) responsible for decommissioning the NEF. In this capacity, LES will have direct control and oversight over all decommissioning activities. The role will be similar to that taken by Urenco at its facilities in Europe. In that role, Urenco has provided operational, technical, licensing, and project management support of identical facilities during both operational and decommissioning campaigns. LES also plans to secure contract services to supplement its capabilities as necessary.

Management of the decommissioning program will assure that proper training and procedures are implemented to assure worker health and safety. Programs and procedures, based on already existing operational procedures, will focus heavily on minimizing waste volumes and worker exposure to hazardous and radioactive materials. Qualified contractors assisting with decommissioning will likewise be subject to facility training requirements and procedural controls.

10.1.5.5 Health and Safety

As with normal operation, the policy during decommissioning shall be to keep individual and collective occupational radiation exposure as low as reasonably achievable (ALARA). A health physics program will identify and control sources of radiation, establish worker protection requirements, and direct the use of survey and monitoring instruments.

10.1.5.6 Waste Management

Radioactive and hazardous wastes produced during decommissioning will be collected, handled, and disposed of in accordance with all regulations applicable to the facility at the time of decommissioning. Generally, procedures will be similar to those described for wastes produced during normal operation. These wastes will ultimately be disposed of in licensed radioactive or hazardous waste disposal facilities located elsewhere. Non-hazardous and non-radioactive wastes will be disposed of consistent with good industrial practice, and in accordance with applicable regulations.

10.1.5.7 Security/Material Control

Requirements for physical security and for material control and accounting will be maintained as required during decommissioning in a manner similar to the programs in force during operation. The LES plan for completion of decommissioning, submitted near the end of plant life, will provide a description of any necessary revisions to these programs.

10.1.5.8 Record Keeping

Records important for safe and effective decommissioning of the facility will be stored in the LES Records Management System until the site is released for unrestricted use. Information maintained in these records includes:

1. Records of spills or other unusual occurrences involving the spread of contamination in and around the facility, equipment, or site. These records may be limited to instances when contamination remains after any cleanup procedures or when there is reasonable likelihood that contaminants may have spread to inaccessible areas as in the case of possible seepage into porous materials such as concrete. These records will include any known information on identification of involved nuclides, quantities, forms, and concentrations.

2. As-built drawings and modifications of structures and equipment in restricted areas where radioactive materials are used and/or stored and of locations of possible inaccessible contamination such as buried pipes which may be subject to contamination. Required drawings will be referenced as necessary, although each relevant document will not be indexed individually. If drawings are not available, appropriate records of available information concerning these areas and locations will be substituted.

3. Except for areas containing only sealed sources, a list contained in a single document and updated every two years, of the following:
   - (i) All areas designed and formerly designated as Restricted Areas as defined under 10 CFR 20.1003; (CFR, 2003c)
   - (ii) All areas outside of Restricted Areas that require documentation specified in item 1 above;

(iii) All areas outside of Restricted Areas where current and previous wastes have been buried as documented under 10 CFR 20.2108 (CFR, 2003d); and

(iv) All areas outside of Restricted Areas that contain material such that, if the license expired, the licensee would be required to either decontaminate the area to meet the criteria for decommissioning in 10 CFR 20, subpart E, (CFR, 2003e) or apply for approval for disposal under 10 CFR 20.2002 (CFR, 2003f).

4. Records of the cost estimate performed for the decommissioning funding plan or of the amount certified for decommissioning, and records of the funding method used for assuring funds if either a funding plan or certification is used.

10.1.6 Decommissioning Process

10.1.6.1 Overview

Implementation of the DECON alternative for decommissioning may begin immediately following Separations Building Module equipment shutdown, since only low radiation levels exist at this facility. In the phased approach presented herein, dismantling and decontamination of the equipment in the three Separations Building Modules will be conducted sequentially (in three phases) over a nine year time frame. Separations Building Module 1 will be decommissioned during the first three year period, followed by Separations Building Module 2 in the next three years, and then Separations Building Module 3 in the final three years. Termination of Separations Building Module 3 operations will mark the end of uranium enrichment operations at the facility. Decommissioning of the remaining plant systems and buildings will begin after Separations Building Module 3 operations have been permanently terminated. A schematic of the NEF decommissioning schedule is presented in Figure 10.1-1, NEF – Conceptual Decommissioning Schedule.

Prior to beginning decommissioning operations, an extensive radiological survey of the facility will be performed in conjunction with a historical site assessment. The findings of the radiological survey and historical site assessment will be presented in a Decommissioning Plan to be submitted to the NRC. The Decommissioning Plan will be prepared in accordance with 10 CFR 70.38 (CFR, 2003a) and the applicable guidance provided in NUREG-1757 (NRC, 2003).

Decommissioning activities will generally include (1) installation of decontamination facilities, (2) purging of process systems, (3) dismantling and removal of equipment, (4) decontamination and destruction of Confidential and Secret Restricted Data material, (5) sales of salvaged materials, (6) disposal of wastes, and (7) completion of a final radiation survey. Credit is not taken for any salvage value that might be realized from the sale of potential assets (e.g., recovered materials or decontaminated equipment) during or after decommissioning.

Decommissioning, using the DECON approach, requires residual radioactivity to be reduced below specified levels so the facilities may be released for unrestricted use. Current Nuclear Material Safety and Safeguards guidelines for release serve as the basis for decontamination costs estimated herein. Portions of the facility that do not exceed contamination limits may remain as is without further decontamination measures applied. The intent of decommissioning the facility is to remove all enrichment-related equipment from the buildings such that only the building shells and site infrastructure remain. The removed equipment includes all piping and components from systems providing UF_6 containment, systems in direct support of enrichment (such as refrigerant and chilled water), radioactive and hazardous waste handling systems, contaminated HVAC filtration systems, etc. The remaining site infrastructure will include services such as electrical power supply, treated water, fire protection, HVAC, cooling water and communications.

Decontamination of plant components and structures will require installation of two new facilities dedicated for that purpose. Existing plant buildings, such as the Centrifuge Assembly Building, are assumed to house the facilities. These facilities will be specially designed to accommodate repetitive cleaning of thousands of centrifuges, and to serve as a general-purpose facility used primarily for cleaning larger components. The two new facilities will be the primary location for decontamination activities during the decommissioning process. The small decontamination area in the Technical Services Building (TSB), used during normal operation, may also handle small items at decommissioning.

Decontaminated components may be reused or sold as scrap. All equipment that is to be reused or sold as scrap will be decontaminated to a level at which further use is unrestricted. Materials that cannot be decontaminated will be disposed of in a licensed radioactive waste disposal facility. As noted earlier, credit is not taken for any salvage value that might be realized from the sale of potential assets (e.g., recovered materials or decontaminated equipment) during or after decommissioning.

Any UF_6 tails remaining on site will be removed during decommissioning. Depending on technological developments occurring prior to plant shutdown, the tails may have become marketable for further enrichment or other processes. The disposition of UF_6 tails and relevant funding provisions are discussed in Section 10.3, Tails Disposition. The cost estimate takes no credit for any value that may be realized in the future due to the potential marketability of the stored tails.

Contaminated portions of the buildings will be decontaminated as required. Structural contamination should be limited to structures in the Restricted Areas. The liners and earthen covers on the facility evaporative basins are assumed to be mildly contaminated and provisions are made for appropriate disposal of these materials in the decommissioning cost estimate. Good housekeeping practices during normal operation will maintain the other areas of the site clean.

When decontamination is complete, all areas and facilities on the site will be surveyed to verify that further decontamination is not required. Decontamination activities will continue until the entire site is demonstrated to be suitable for unrestricted use.

10.1.6.2 Decontamination Facility Construction

New facilities for decontamination can be installed in existing plant buildings to avoid unnecessary expense. Estimated time for equipment installation is approximately one year. These new facilities will be completed in time to support the dismantling and decontamination of Separations Building Module 1. These facilities are described in Section 10.1.7, Decontamination Facilities.

10.1.6.3 System Cleaning

At the end of the useful life of each Separations Building Module, the enrichment process is shut down and UF_6 is removed to the fullest extent possible by normal process operation. This is followed by evacuation and purging with nitrogen. This shutdown and purging portion of the decommissioning process is estimated to take approximately three months.

10.1.6.4 Dismantling

Dismantling is simply a matter of cutting and disconnecting all components requiring removal. The operations themselves are simple but very labor intensive. They generally require the use of protective clothing. The work process will be optimized, considering the following.

- Minimizing the spread of contamination and the need for protective clothing
- Balancing the number of cutting and removal operations with the resultant decontamination and disposal requirements
- Optimizing the rate of dismantling with the rate of decontamination facility throughput
- Providing storage and laydown space required, as impacted by retrievability, criticality safety, security, etc
- Balancing the cost of decontamination and salvage with the cost of disposal.

Details of the complex optimization process will necessarily be decided near the end of plant life, taking into account specific contamination levels, market conditions, and available waste disposal sites. To avoid laydown space and contamination problems, dismantling should be allowed to proceed generally no faster than the downstream decontamination process. The time frame to accomplish both dismantling and decontamination is estimated to be approximately three years per Separations Building Module.

10.1.6.5 Decontamination

The decontamination process is addressed separately in detail in Section 10.1.7.

10.1.6.6 Salvage of Equipment and Materials

Items to be removed from the facilities can be categorized as potentially re-usable equipment, recoverable scrap, and wastes. However, based on a 30 year facility operating license, operating equipment is not assumed to have reuse value. Wastes will also have no salvage value.

With respect to scrap, a significant amount of aluminum will be recovered, along with smaller amounts of steel, copper, and other metals. For security and convenience, the uncontaminated materials will likely be smelted to standard ingots, and, if possible, sold at market price. The contaminated materials will be disposed of as low-level radioactive waste. No credit is taken for any salvage value that might be realized from the sale of potential assets during or after decommissioning.

10.1.6.7 Disposal

All wastes produced during decommissioning will be collected, handled, and disposed of in a manner similar to that described for those wastes produced during normal operation. Wastes will consist of normal industrial trash, non-hazardous chemicals and fluids, small amounts of hazardous materials, and radioactive wastes. The radioactive waste will consist primarily of crushed centrifuge rotors, trash, and citric cake. Citric cake consists of uranium and metallic compounds precipitated from citric acid decontamination solutions. It is estimated that approximately 5,000 m^3 (6,539 yd^3) of radioactive waste will be generated over the nine-year decommissioning operations period. (This waste is subject to further volume reduction processes prior to disposal).

Radioactive wastes will ultimately be disposed of in licensed low-level radioactive waste disposal facilities. Hazardous wastes will be disposed of in hazardous waste disposal facilities. Non-hazardous and non-radioactive wastes will be disposed of in a manner consistent with good industrial practice and in accordance with all applicable regulations. A complete estimate of the wastes and effluent to be produced during decommissioning will be provided in the Decommissioning Plan that will be submitted prior to initiating the decommissioning of the plant.

Confidential and Secret Restricted Data components and documents on site shall be disposed of in accordance with the requirements of 10 CFR 95 (CFR, 2003g). Such classified portions of the centrifuges will be destroyed, piping will likely be smelted, documents will be destroyed, and other items will be handled in an appropriate manner. Details will be provided in the facility Standard Practice Procedures Plan for the Protection of Classified Matter and Information, submitted separately in accordance with 10 CFR 95 (CFR, 2003g).

10.1.6.8 Final Radiation Survey

A final radiation survey must be performed to verify proper decontamination to allow the site to be released for unrestricted use. The evaluation of the final radiation survey is based in part on an initial radiation survey performed prior to initial operation. The initial survey determines the natural background radiation of the area; therefore it provides a datum for measurements which determine any increase in levels of radioactivity.

The final survey will systematically measure radioactivity over the entire site. The intensity of the survey will vary depending on the location (i.e. the buildings, the immediate area around the buildings, and the remainder of the site). The survey procedures and results will be documented in a report. The report will include, among other things, a map of the survey site, measurement results, and the site's relationship to the surrounding area. The results will be analyzed and shown to be below allowable residual radioactivity limits; otherwise, further decontamination will be performed.

10.1.7 Decontamination Facilities

10.1.7.1 Overview

The facilities, procedures, and expected results of decontamination are described in the paragraphs below. Since reprocessed uranium will not be used as feed in the NEF, no consideration of ^{232}U, transuranic alpha-emitters and fission product residues is necessary for the decontamination process. Only contamination from ^{238}U, ^{235}U, ^{234}U, and their daughter products will require handling by decontamination processes. The primary contaminant throughout the plant will be in the form of small amounts of UO_2F_2, with even smaller amounts of UF_4 and other compounds.

10.1.7.2 Facilities Description

A decontamination facility will be required to accommodate decommissioning. This specialized facility is needed for optimal handling of the thousands of centrifuges to be decontaminated, along with the UF_6 vacuum pumps and valves. Additionally, a general purpose facility is required for handling the remainder of the various plant components. These facilities are assumed to be installed in existing plant buildings (such as the Centrifuge Assembly Building).

The decontamination facility will have four functional areas that include (1) a disassembly area, (2) a buffer stock area, (3) a decontamination area, and (4) a scrap storage area for cleaned stock. The general purpose facility may share the specialized decontamination area. However, due to various sizes and shapes of other plant components needing handling, the disassembly area, buffer stock areas and scrap storage areas may not be shared. Barriers and other physical measures will be installed and administrative controls implemented, as needed, to limit the spread of contamination.

Equipment in the decontamination facility is assumed to include:

- Transport and manipulation equipment
- Dismantling tables for centrifuge externals
- Sawing machines

- Dismantling boxes and tanks, for centrifuge internals
- Degreasers
- Citric acid and demineralized water baths
- Contamination monitors
- Wet blast cabinets
- Crusher, for centrifuge rotors
- *Smelting and/or shredding equipment*
- Scrubbing facility.

The decontamination facilities provided in the TSB for normal operational needs would also be available for cleaning small items during decommissioning.

10.1.7.3 Procedures

Formal procedures for all major decommissioning activities will be developed and approved by plant management to minimize worker exposure and waste volumes, and to assure work is carried out in a safe manner. The experience of decommissioning European gas centrifuge enrichment facilities will be incorporated extensively into the procedures.

At the end of plant life, some of the equipment, most of the buildings, and all of the outdoor areas should already be acceptable for release for unrestricted use. If they are accidentally contaminated during normal operation, they would be cleaned up when the contamination is discovered. This limits the scope of necessary decontamination at the time of decommissioning.

Contaminated plant components will be cut up or dismantled, then processed through the decontamination facilities. Contamination of site structures will be limited to areas in the Separations Building Modules and TSB, and will be maintained at low levels throughout plant operation by regular cleaning. The Decontamination Workshop Area, Ventilated Room, Vacuum Pump Rebuild Workshop, and a portion of the Laundry Room are included as permanent Restricted Areas. Through the application of special protective coatings, to surfaces that might become radioactively contaminated during operation, and good housekeeping practices, final decontamination of these areas is assumed to require minimal removal of surface concrete or other structural material.

The centrifuges will be processed through the specialized facility. The following operations will be performed.

- Removal of external fittings
- Removal of bottom flange, motor and bearings, and collection of contaminated oil

- Removal of top flange, and withdrawal and disassembly of internals
- Degreasing of items as required
- Decontamination of all recoverable items for smelting
- Destruction of other classified portions by shredding, crushing, smelting, etc.

10.1.7.4 Results

Urenco plant experience in Europe has demonstrated that conventional decontamination techniques are effective for all plant items. Recoverable items have been decontaminated and made suitable for reuse except for a very small amount of intractably contaminated material. The majority of radioactive waste requiring disposal in the NEF will include crushed centrifuge rotors, trash, and residue from the effluent treatment systems.

European experience has demonstrated that the aluminum centrifuge casings can be successfully decontaminated and recycled. However, as a conservative measure for this decommissioning cost estimate, the aluminum centrifuge casings for the NEF are assumed to be disposed of as low-level radioactive waste.

Overall, no problems are anticipated that will prevent the site from being released for unrestricted use.

10.1.7.5 Decommissioning Impact on Integrated Safety Analysis (ISA)

As was described in Section 10.1.3.1, Summary of Costs, dismantling and decontamination of the equipment in the three Separations Building Modules will be conducted sequentially (in three phases) over a nine year time frame. Separations Building Module 1 will be decommissioned during the first three-year period, followed by Separations Building Module 2, and then Separations Building Module 3. Termination of Separations Module 3 operations will mark the end of uranium enrichment operations at the NEF. Decommissioning of the remaining plant systems and buildings will begin after Separations Building Module 3 operations have been permanently terminated.

Although decommissioning operations are planned to be underway while all the activities considered in the ISA continue to occur in the other portions of the plant, the current ISA has not considered these decommissioning risks. An updated ISA will be performed at a later date, but prior to decommissioning, to incorporate the risks from decommissioning operations on concurrent enrichment operations.

10.2 FINANCIAL ASSURANCE MECHANISM

10.2.1 Decommissioning Funding Mechanism

LES intends to utilize a surety method to provide reasonable assurance of decommissioning funding as required by 10 CFR 40.36(e)(2) (CFR, 2003h) and 70.25(f)(2) (CFR, 2003i). *Finalization of the specific financial instruments to be utilized will be completed, and signed* originals of those instruments will be provided to the NRC, prior to LES receipt of licensed material. LES intends to provide continuous financial assurance from the time of receipt of licensed material to the completion of decommissioning and termination of the license. Since LES intends to sequentially install and operate the Separations Building Modules over time, financial assurance for decommissioning will be provided during the operating life of the NEF at a rate that is in proportion to the decommissioning liability for these facilities as they are phased in. Similarly, LES will provide decommissioning funding assurance for disposition of depleted tails at a rate in proportion to the amount of accumulated tails onsite up to the maximum amount of the tails as described in Section 10.3, Tails Disposition. An exemption request to permit this incremental financial assurance is provided in Section 1.2.5, "Special Exemptions or Special Authorizations."

The surety method adopted by LES will provide an ultimate guarantee that decommissioning costs will be paid in the event LES is unable to meet its decommissioning obligations at the time of decommissioning. The surety method will also be structured and adopted consistent with applicable NRC regulatory requirements and in accordance with NRC regulatory guidance contained in NUREG-1757 (NRC, 2003). Accordingly, LES intends that its surety method will contain, but not be limited to, the following attributes:

- The surety method will be open-ended or, if written for a specified term, such as five years, will be renewed automatically unless 90 days or more prior to the renewal date, the issuer notifies the NRC, the trust to which the surety is payable, and LES of its intention not to renew. The surety method will also provide that the full face amount be paid to the beneficiary automatically prior to the expiration without proof of forfeiture if LES fails to provide a replacement acceptable to the NRC within 30 days after receipt of notification of cancellation.

- The surety method will be payable to a trust established for decommissioning costs. The trustee and trust will be ones acceptable to the NRC. For instance, the trustee may be an appropriate State or Federal government agency or an entity which has the authority to act as a trustee and whose trust operations are regulated and examined by a Federal or State agency.

- The surety method will remain in effect until the NRC has terminated the license.

- Unexecuted copies of the surety method documentation are provided in Appendices 10A through 10F. Prior to LES receipt of licensed material, the applicable unexecuted copies of the surety method documentation will be replaced with the finalized, signed, and executed surety method documentation, including a copy of the broker/agent's power of attorney authorizing the broker/agent to issue bonds.

10.2.2 Adjusting Decommissioning Costs and Funding

In accordance with 10 CFR 40.36(d) (CFR, 2003h) and 70.25(e) (CFR, 2003i), LES will update the decommissioning cost estimate for the NEF, and the associated funding levels, over the life of the facility. These updates will take into account changes resulting from inflation or site-specific factors, such as changes in facility conditions or expected decommissioning procedures. These funding level updates will also address anticipated operation of additional Separations Building Modules and accumulated tails.

As required by the applicable regulations 10 CFR 70.25(e) (CFR, 2003i), such updating will occur approximately every three years. A record of the update process and results will be retained for review as discussed in Section 10.2.3, below. The NRC will be notified of any material changes to the decommissioning cost estimate and associated funding levels (e.g., significant increases in costs beyond anticipated inflation). To the extent the underlying instruments are revised to reflect changes in funding levels, the NRC will be notified as appropriate.

In addition to the triennial update of the decommissioning cost estimate described above, LES has committed to supplemental updates as described in the request for exemption in SAR Section 1.2.5 in order to ensure adequate financial assurance on an incremental basis. Specifically, LES commits to update the decommissioning cost estimates and to provide to the NRC a revised funding instrument for facility decommissioning prior to the operation of each Separations Building Module at a minimum. LES also commits to updating the cost estimates for the dispositioning of the depleted uranium byproduct on an annual forward-looking incremental basis and to providing the NRC revised funding instruments that reflect these projections of depleted uranium byproduct production. If any adjustments to the funding assurance are determined to be needed during this annual period due to production variations, they would be made promptly and a revised funding instrument would be provided to the NRC.

For the first triennial period, LES intends to provide decommissioning funding assurance for the entire facility, incorporating the three Separations Building Modules, and the amount of depleted uranium byproduct that would be produced by the end of that first three year period. In 2004 dollars, the following cost estimates would be assured: 1) the total facility decommissioning cost estimate of $131,103,000 from Table 10.1-14, "Total Decommissioning Costs," 2) the cost for dispositioning 4,861 MT of depleted uranium byproduct, the amount produced at the end of the first three years of operation, based on a projected nominal 30 years of operation, and using a cost of $4.68 per kg of depleted uranium byproduct, ($4,680 per MT depleted uranium byproduct) from SAR Section 10.3, yielding a total of $22,749,480, and 3) applying a 25% contingency factor to the total, or $38,463,120. Accordingly the total projected decommissioning cost estimate for the first triennial period of NEF operation for which financial assurance would be provided would be $192,315,600. However, if significant deviations to the facility construction or initial operation schedules are encountered after the first triennial period, LES may instead provide decommissioning funding assurance on the incremental basis described above, i.e., prior to the operation of a Separations Building Module and on an annual basis for the depleted uranium byproduct.

10.2.3 Recordkeeping Plans Related to Decommissioning Funding

In accordance with 10 CFR 40.36(f) (CFR, 2003h) and 70.25(g) (CFR, 2003i), LES will retain records, until the termination of the license, of information that could have a material effect on the ultimate costs of decommissioning. These records will include information regarding: (1) spills or other contamination that cause contaminants to remain following cleanup efforts; (2) as-built drawings of structures and equipment, and modifications thereto, where radioactive contamination exists (e.g., from the use or storage of such materials); (3) original and modified cost estimates of decommissioning; and (4) original and modified decommissioning funding instruments and supporting documentation.

10.3 TAILS DISPOSITION

The disposition of tails from the NEF is an element of authorized operating activities. It involves neither decommissioning waste nor is it a part of decommissioning activities. The disposal of these tails is analogous to the disposal of radioactive materials generated in the course of normal operations (even including spent fuel in the case of a power reactor), which is authorized by the operating license and subject to separate disposition requirements. Such costs are not appropriately included in decommissioning costs (this principle (in the 10 CFR 50 context) is discussed in Regulatory Guide 1.159 (NRC, 1990), Section 1.4.2, page 1.159-8). Further, the "tails" products from the NEF are not mill tailings, as regulated pursuant to the Uranium Mill Tailings Radiation Control Act, as amended and 10 CFR 40, Appendix A (CFR, 2003j), and are not subject to the financial requirements applicable to mill tailings.

Nevertheless, LES intends to provide for expected tails disposition costs (even assuming ultimate disposal as waste) during the life of the facility. Funds to cover these costs are based on the amount of tails generated and the unit cost for the disposal of depleted UF_6.

It is anticipated that the NEF will generate 132,942 MT of depleted uranium over a nominal 30 year operational period. This estimate is conservative as it assumes continuous production of tails over 30 years of operation. Actual tails production will cease prior to the end of the license term as shown in Figure 10.1-1, NEF – Conceptual Decommissioning Schedule.

Waste processing and disposal costs for UF_6 tails are currently estimated to be \$5.50 per kg U or \$5,500 per MT U. This unit cost was obtained from four sets of cost estimates for the conversion of DUF_6 to DU_3O_8 and the disposal of DU_3O_8 product, and the transportation of DUF_6 and DU_3O_8. The cost estimates were obtained from analyses of four sources: a 1997 study by the Lawrence Livermore National Laboratory (LLNL) (Elayat, 1997), the Uranium Disposition Services (UDS) contract with the Department of Energy (DOE) of August 29, 2002 (DOE, 2002), information from Urenco, and the costs submitted to the Nuclear Regulatory Commission as part of the Claiborne Enrichment Center (CEC) license application (LES, 1993a) in the 1990s.

The four sets of cost estimates obtained are presented in Table 10.3-1, Summary Of Depleted UF_6 Disposal Costs From Four Sources, below, in 2002 dollars per kg of uranium (kg U). Note that the Claiborne Energy Center cost had a greater uncertainty associated with it. The UDS contract does not allow the component costs for conversion, disposal and transportation to be estimated. The costs in the table indicate that \$5.50 per kg U (\$2.50 per lb U) is a conservative and, therefore, prudent estimate of total depleted UF_6 disposition cost for the LES NEF. That is, the historical cost estimates from LLNL and CEC and the more recent actual costs from the UDS contract were used to inform the LES cost estimate. Urenco has reviewed this estimate and, based on its current cost for UBC disposal, finds this figure to be prudent.

In May 1997, the LLNL published UCRL-AR-127650, Cost Analysis Report for the Long-Term Management of Depleted Uranium Hexafluoride (Elayat, 1997). The report was prepared to provide comparative life-cycle cost data for the Department of Energy's (DOE's) Draft 1997 Programmatic Environmental Impact Statement (PEIS) (DOE, 1997) on alternative strategies for management and disposition of DUF_6. The LLNL report is the most comprehensive assessment of DUF_6 disposition costs for alternative disposition strategies available in the public domain.

The technical data on which the LLNL report is based is principally the May 1997 Engineering Analysis Report (UCRL-AR-124080, Volumes 1 and 2) (Dubrin, 1997).

When the LLNL report was prepared in 1997, more than six years ago, the cost estimates in it were based on an inventory of 560,000 MT of DUF_6, or 378,600 MTU after applying the 0.676 mass fraction multiplier. This amount corresponds to an annual throughput rate of 28,000 MT of UF6 or about 19,000 MTU of depleted uranium. The costs in the LLNL report are based on the 20 year life-cycle quantity of 378,600 MTU. The LLNL annual DUF_6 quantities are about 3.6 times the annual production rate of the proposed NEF.

The LLNL cost analyses assumed that the DUF_6 would be converted to DU_3O_8, the DOE's preferred disposal form, using one of two dry process conversion options. The first --- the anhydrous hydrogen fluoride (AHF) option ---- upgrades the hydrogen fluoride (HF) product to anhydrous HF (< 1.0% water). In the second option --- the HF neutralization option --- the hydrofluoric acid would be neutralized with lime to produce calcium fluoride (CaF_2). The LLNL cost analyses assumed that the AHF and CaF_2 conversion products are of sufficient purity that they could be sold for unrestricted use (negligible uranium contamination). LES will not use a deconversion facility that employs a process that results in the production of anhydrous HF.

The costs in Table 10.3-1, represent the LLNL-estimated life-cycle capital, operating, and regulatory costs, in 2002 dollars, for conversion of 378,600 MTU over 20 years, of DUF_6 to DU_3O_8 by anhydrous hydrogen fluoride (HF) processing, followed by DU_3O_8 long-term storage disposal in a concrete vault, or in an exhausted underground uranium mine in the western United States, at or below the same cost. An independent new underground mine production cost analysis confirmed that the LLNL concrete vault alternative costs represent an upper bound for under ground mine disposal. The discounted 1996 dollar costs in the LLNL report were undiscounted and escalated to 2002 dollars. The LLNL life-cycle costs in 1996 dollars were converted to per kgU costs and adjusted to 2002 dollars using the Gross Domestic Product (GDP) Implicit Price Deflator (IPD). The escalation adjustment resulted in the 1996 costs being escalated by 11%.

On August 29, 2002, the DOE announced the competitive selection of Uranium Disposition Services, LLC to design, construct, and operate conversion facilities near the DOE enrichment plants at Paducah, Kentucky and Portsmouth, Ohio. UDS will operate these facilities for the first five years, beginning in 2005. The UDS contract runs from August 29, 2002 to August 3, 2010. UDS will also be responsible for maintaining the depleted uranium and product inventories and transporting depleted uranium from Oak Ridge East Tennessee Technology Park (ETTP) to the Portsmouth site for conversion. The DOE-UDS contract scope includes packaging, transporting and disposing of the conversion product DU_3O_8.

UDS is a consortium formed by Framatome ANP Inc., Duratek Federal Services Inc., and Burns and Roe Enterprises Inc. The DOE-estimated value of the cost reimbursement contract is $558 million (DOE Press Release, August 29, 2002) (DOE, 2002). Design, construction and operation of the facilities will be subject to appropriations of funds from Congress. On December 19, 2002, the White House confirmed that funding for both conversion facilities will be included in President Bush's 2004 budget. However, the Office of Management and Budget has not yet indicated how much funding will be allocated. The UDS contract quantities and costs are given in Table 10.3-2, DOE-UDS August 29, 2002, Contract Quantities and Costs.

Urenco is currently contracted with a supplier for DUF_6 to DU_3O_8 conversion. The supplier has been converting DUF_6 to DU_3O_8 on an industrial scale since 1984.

The CEC costs given in Table 10.3-1, are those presented to John Hickey of the NRC in the CEC letter of June 30, 1993 (LES, 1993b) as adjusted for changes in units and escalated to 2002 ($6.74 per kgU). The conversion cost of $4.00 per kg U was provided to CEC by Cogema at that time. It should also be noted that this highest cost estimate is at least 10 years old and was based on the information available at that time. The value of $5.50 per kgU used in the decommissioning cost estimate is 22% above the average of the more recent LLNL and UDS cost estimates, which is $4.49 per kgU {(5.06+3.92)/2}. The LLNL Cost Analysis Report (page 30) states that its cost estimate already includes a 30% contingency in the capital costs of the process and manufacturing facilities, a 20% contingency in the capital costs of the balance of plant; and a minimum of a 30% contingency in the capital costs of process and manufacturing equipment.

Also, the 1997 LLNL cost information is five years older than the more recent 2002 UDS cost information. The value of $5.50 per kgU used in the decommissioning cost estimate for tails disposition is 40% greater than the 2002 UDS-based cost estimate of $3.92 per kgU, which does not include offset credits for HF sales or proceeds from the sale of recycled products.

The costs in Table 10.3-1, indicate that $5.50 is a conservative and, therefore, prudent estimate of total DU disposition cost for the NEF. Urenco has reviewed this estimate and, based on its current cost after tails disposal, finds this figure to be prudent.

In summary, there is already substantial margin between the value of $5.50 per kgU being used by LES in the decommissioning cost estimate and the most recent information (2002 UDS) from which LES derived a cost estimate of $3.92 per kgU.

Based on information from corresponding vendors, the value of $5.50 per kgU (2002 dollars), which is equal to $5.70 per kgU when escalated to 2004 dollars, was revised in December 2004 to $4.68 per kgU (2004 dollars). The value of $4.68 per kgU was derived from the estimates of costs from the three components that make up the total disposition cost of DUF_6 (i.e., deconversion, disposal, and transportation). The estimate of $4.68 per kgU supports the Preferred Plausible Strategy of U.S. Private Sector Conversion and Disposal identified in section 4.13.3.1.3 of the ER as Option 1.

In support of the Option 2 Plausible Strategy identified in section 4.13.3.1.3 of the ER, "DOE Conversion and Disposal," LES requested a cost estimate from the Department of Energy (DOE). On March 1, 2005, DOE provided a cost estimate to LES for the components that make up the total disposition cost (i.e., deconversion, disposal, and transportation) (DOE, 2005). This estimate, which was based upon an independent analysis undertaken by DOE's consultant, LMI Government Consulting, estimated the cost of disposition to total approximately $4.91 per kgU (2004 dollars). The Department's cost estimate for deconversion, storage, and disposal of the DU is consistent with the contract between UDS and DOE. The cost estimate does not assume any resale or reuse of any products resulting from the conversion process.

For purposes of determining the total tails disposition funding requirement and the amount of financial assurance required for this purpose, the value of $4.68 per kgU (based upon the cost estimate for the Preferred Plausible Strategy) was selected. Based on a computed tails production of 132,942 MTU during a nominal 30 years of operation and a tails processing cost of $4.68 per kgU or $4,680 per MTU, the total tails disposition funding requirement is estimated at $622,169,000. This sum will be included as part of the financial assurance for decommissioning (see Table 10.1-14, Total Decommissioning Costs). See Environmental Report Section 4.13.3.1.6, Costs Associated with UF_6 Tails Conversion and Disposal, for additional details.

10.4 REFERENCES

CFR, 2003a. Title 10, Code of Federal Regulations, Section 70.38, Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Section 20.1402, Radiological criteria for unrestricted use, 2003.

CFR, 2003c. Title 10, Code of Federal Regulations, Part 20.1003, Definitions, 2003.

CFR, 2003d. Title 10, Code of Federal Regulations, Part 20.2108, Records of waste disposal, 2003.

CFR, 2003e. Title 10, Code of Federal Regulations, Part 20, Subpart E, Radiological Criteria for License Termination, 2003.

CFR, 2003f. Title 10, Code of Federal Regulations, Part 20.2002, Method for obtaining approval of proposed disposal procedures, 2003.

CFR, 2003g. Title 10, Code of Federal Regulations, Part 95, Security Facility Approval and Safeguarding of National Security Information and Restricted Data, 2003.

CFR, 2003h. Title 10, Code of Federal Regulations, Section 40.36, Financial assurance and recordkeeping for decommissioning, 2003.

CFR, 2003i. Title 10, Code of Federal Regulations, Section 70.25, Financial assurance and recordkeeping for decommissioning, 2003.

CFR, 2003j. Title 10, Code of Federal Regulations, Part 40, Appendix A, Criteria Relating to the Operation of Uranium Mills and the Disposition of Tailings or Wastes Produced by the Extraction or Concentration of Source Material From Ores Processed Primarily for Their Source Material Content, 2003.

DOE, 1997. Programmatic Environmental Impact Statement for Alternative Strategies for the Long-Term Management and Use of Depleted Uranium Hexafluoride, U.S. Department of Energy, December 1997.

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Dubrin, 1997. "Depleted Uranium Hexafluoride Management Program", UCRL-AR-124080 Vol. 1 Rev. 2 and Vol. 2, Lawrence Livermore National Laboratory, Dubrin, J.W., et. al., May 1997.

Elayat, 1997. "Cost Analysis Report For the Long-Term Management of Depleted Uranium Hexafluoride", UCRL-AR-127650, Lawrence Livermore National Laboratory, Elayat, Hatem, J.Zoller, L. Szytel, May 1997.

LES, 1993a. Clairborne Enrichment Center Safety Analysis Report, Section 11.8, Decommissioning, Louisiana Energy Services, 1993.

LES, 1993b. Letter from Peter G. LeRoy, Louisiana Energy Services, to John W.N. Hickey, U.S. Nuclear Regulatory Commission, June 30, 1993.

NRC, 1990. Assuring the Availability of Funds for Decommissioning Nuclear Reactors, Regulatory Guide 1.159, U.S. Nuclear Regulatory Commission, August 1990.

NRC, 1994. Safety Evaluation Report for the Claiborne Enrichment Center, Homer, Louisiana, NUREG-1491, U.S. Nuclear Regulatory Commission, January 1994.

NRC, 2003. Consolidated NMSS Decommissioning Guidance – Financial Assurance, Recordkeeping, and Timeliness, NUREG-1757, Volume 3, U.S. Nuclear Regulatory Commission, September 2003.

TABLES

Table 10.1-1A Number and Dimensions of Facility Components

Separations Modules (Note 1)

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes			
Fume Cupboards			
Lab Benches			
Sinks			
Drains			
Floors			
Walls			
Ceilings			
Ventilation/Ductwork			
Hot Cells			
Equipment/Materials			
Soil Plots			
Storage Tanks			
Storage Areas			
Radwaste Areas			
Scrap Recovery Areas			
Maintenance Shop			
Equipment Decontamination Areas			
Other			

Notes:

1. More than 97% of the decommissioning costs for the facility are attributed to the dismantling, decontamination, processing, and disposal of centrifuges and other equipment in the Separations Building Modules, which are considered classified. Given the classified nature of these buildings, the data presented in these Tables have been structured to meet the applicable NUREG-1757 recommendations, to the extent practicable. However, specific information regarding numbers of components, dimensions of components, and total dimensions, has been intentionally excluded to protect the classified nature of the data.

Table 10.1-1B Number and Dimensions of Facility Components
Page 1 of 1

Decommission Decontamination Facility

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes	None	NA	NA
Fume Cupboards	None	NA	NA
Lab Benches	10	Various sizes of lab and workshop benches ranging from 6.5 to 13 feet long by 2.5 feet wide	(Note 1)
Sinks	6	Standard laboratory sinks and hand wash basins	(Note 1)
Drains	6	Standard laboratory type drains	(Note 1)
Floors	1 Lot (Note 2)	(Note 1)	(Note 1)
Walls	1 Lot (Note 2)	(Note 1)	(Note 1)
Ceilings	1 Lot (Note 2)	(Note 1)	(Note 1)
Ventilation/Ductwork	(Note 3)	Various sizes of ductwork ranging from 3 to 18 inches plus dampers, valves and flexibles	640 feet
Hot Cells	None	NA	NA
Equipment/Materials	20	Various pieces of equipment including citric cleaning tanks, centrifuge cutting machines	(Note 1)
Soil Plots	None	NA	NA
Storage Tanks	1 Lot (Note 2)	Various storage tanks	(Note 1)
Storage Areas	1	Storage area for centrifuges and pipe work	(Note 1)
Radwaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	None	NA	NA
Equipment Decontamination Areas	None	NA	NA
Other	1 Lot (Note 2)	Hand tools and consumables that become contaminated while carrying out dismantling and decontamination work, unmeasured work and scaffolding	(Note 1)

Notes:

1. Total dimensions not used in estimating model.
2. Allocation based on Urenco decommissioning experience.
3. Total dimensions provided.

Table 10.1-1C Number and Dimensions of Facility Components
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Technical Services Building

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes	None	NA	NA
Fume Cupboards	18	Standard laboratory fume cupboards, approx 6.5 - 8 feet high x 5 feet wide	(Note 1)
Lab Benches	25	Various sizes of lab and workshop benches ranging from 6.5 – 13 feet long by 2.5 feet wide	(Note 1)
Sinks	12	Standard laboratory sinks and hand wash basins plus larger sinks for laundry	(Note 1)
Drains	12	Standard Laboratory type drains plus larger laundry drain	(Note 1)
Floors	(Note 3)	Floor area covers all Workshops and Labs in the Technical Services Bldg that may be exposed to contamination	26,340 ft^2
Walls	(Note 3)	Wall area covers all Workshops and Labs in the Technical Services Bldg that may be exposed to contamination	40,074 ft^2
Ceilings	(Note 3)	Ceiling area covers all Workshops and Labs in the Technical Services Bldg that may be exposed to contamination	26,340 ft^2
Ventilation/ Ductwork	(Note 3)	Various pieces of equipment including, filter banks, extractor fans, vent stack, dampers and approx 2,034 feet of large and small ductwork	2,034 feet
Hot Cells	None	NA	NA
Equipment/ Materials	57	Various pieces of equipment including, mass spectrometers, washing machines, hydraulic lift tables, cleaning cabinets	(Note 1)
Soil Plots	None	NA	NA
Storage Tanks	1	Waste oil storage tank (53 gal)	(Note 1)
Storage Areas	2	Storage area for product removal, dirty pumps	(Note 1)
Radwaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	None	NA	NA
Equipment Decontamination Areas	None	NA	NA
Other	1 Lot (Note 2)	Hand tools and consumables that become contaminated while carrying out dismantling/decontamination work, unmeasured work and scaffolding	(Note 1)

Notes:

1. Total dimensions not used in estimating model.
2. Allocation based on Urenco decommissioning experience.
3. Total dimensions provided.

Table 10.1-1D Number and Dimensions of Facility Components

Gaseous Effluent Vent (GEV) System Throughout Plant

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes	None	NA	NA
Fume Cupboards	None	NA	NA
Lab Benches	None	NA	NA
Sinks	None	NA	NA
Drains	None	NA	NA
Floors	None	NA	NA
Walls	None	NA	NA
Ceilings	None	NA	NA
Ventilation/Ductwork	(Note 3)	Various sizes of ductwork ranging from 3 to 18 inches plus dampers, valves and flexibles	5,656 feet
Hot Cells	None	NA	NA
Equipment/Materials	None	NA	NA
Soil Plots	None	NA	NA
Storage Tanks	None	NA	NA
Storage Areas	None	NA	NA
RadWaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	None	NA	NA
Equipment Decontamination Areas	None	NA	NA
Other	1 Lot (Note 2)	Hand tools and consumables that become contaminated while carrying out dismantling/decontamination work, unmeasured work and scaffolding	(Note 1)

Notes:

1. Total dimensions not used in estimating model.
2. Allocation based on Urenco decommissioning experience.
3. Total dimensions provided.

Table 10.1-1E Number and Dimensions of Facility Components

Blending and Sampling

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes	None	NA	NA
Fume Cupboards	None	NA	NA
Lab Benches	None	NA	NA
Sinks	None	NA	NA
Drains	None	NA	NA
Floors	None (Note 4)	NA	NA
Walls	None (Note 4)	NA	NA
Ceilings	None (Note 4)	NA	NA
Ventilation/Ductwork	Covered in GEV System estimate	Covered in GEV System estimate	Covered in GEV System estimate
Hot Cells	None	NA	NA
Equipment/Materials	(Note 3)	Various sizes of pipe-work ranging from DN25 to DN65	2,461 feet
	38 Valves	Various types of valve ranging from 0.6 to 2.5 inches and manual to control	(Note 1)
	12	Various pieces of equipment including hot boxes and traps	(Note 1)
Soil Plots	None	NA	NA
Storage Tanks	None	NA	NA
Storage Areas	None	NA	NA
Radwaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	None	NA	NA
Equipment Decontamination Areas	None	NA	NA
Other	1 Lot (Note 2)	Hand tools and consumables that become contaminated while carrying out dismantling/decontamination work, unmeasured work and scaffolding	(Note 1)

Notes:

1. Total dimensions not used in estimating model.
2. Allocation based on Urenco decommissioning experience.
3. Total dimensions provided.
4. No floors, walls or ceilings are anticipated needing decontamination.

Table 10.1-1F Number and Dimensions of Facility Components
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Centrifuge Test and Post Mortem

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes	None	NA	NA
Fume Cupboards	None	NA	NA
Lab Benches	4	Various sizes of lab and workshop benches ranging from 6.5 – 13 feet long by 2.5 feet wide	(Note 1)
Sinks	2	Standard laboratory sinks and hand wash basins plus larger sinks for laundry	(Note 1)
Drains	2	Standard laboratory type drains plus larger laundry drain	(Note 1)
Floors	None (Note 4)	NA	NA
Walls	None (Note 4)	NA	NA
Ceilings	None (Note 4)	NA	NA
Ventilation/ Ductwork	None	NA	NA
Hot Cells	None	NA	NA
Equipment/ Materials	(Note 3)	Various sizes of pipe-work ranging from DN16 to DN40	164 feet
	56 Valves	Various types of valve ranging from 0.6 to 1.6 inches and manual to control	(Note 1)
	7	Various pieces of equipment including feed take off vessels and traps	(Note 1)
Soil Plots	None	NA	NA
Storage Tanks	None	NA	NA
Storage Areas	None	NA	NA
Radwaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	None	NA	NA
Equipment Decontamination Areas	None	NA	NA
Other	1 Lot (Note 2)	Hand tools and consumables that become contaminated while carrying out dismantling/decontamination work, unmeasured work and scaffolding	(Note 1)

Notes:

1. Total dimensions not used in estimating model.
2. Allocation based on Urenco decommissioning experience.
3. Total dimensions provided.
4. No floors, walls or ceilings are anticipated needing decontamination.

Table 10.1-2 Planning and Preparation
Page 1 of 1

Activity	Costs ($000)	Labor Shift-worker (multi-functional) (Man-days)	Labor Project Management (Man-days)	Labor HP&S (Man-days)	Activity Duration (Months)
Project Plan & Schedule	100	0	178	0	4
Site Characterization Plan	200	0	356	0	4
Site Characterization	300	82	368	144	4
Decommissioning Plan	350	0	622	0	6
NRC Review Period	50	0	89	0	12
Site Services Specifications	100	0	178	0	2
Project Procedures	100	0	178	0	4
TOTAL	1,200	82	1,969	144	(Note 1)

Note:

1. Some activities will be conducted in parallel to achieve a 24 month time frame.

Table 10.1-3 Decontamination or Dismantling of Radioactive Components (Man-Hours)

Page 1 of 1

Other Buildings (Note 1)

Component	Decon Method (Note 4)	Craftsman	Supervision (Note 2)	Project Management	HP&S/Chem (Note 3)
Glove Boxes		0	0	0	0
Fume Cupboards		312	62	53	66
Lab Benches		324	64	55	68
Sinks		101	20	17	21
Drains		102	20	17	21
Floors		647	129	111	136
Walls		422	84	72	89
Ceilings		275	55	47	58
Ventilation/Ductwork		8,468	1,693	1,447	1,780
Hot Cells		0	0	0	0
Equipment/Materials		1,533	307	262	322
Soil Plots		0	0	0	0
Storage Tanks		14	3	2	3
Storage Areas		110	22	19	23
Radwaste Areas		0	0	0	0
Scrap Recovery Areas		0	0	0	0
Maintenance Shop		0	0	0	0
Equipment Decontamination Areas		0	0	0	0
Other		1,913	382	327	402
TOTAL Hours	–	14,221	2,841	2,430	2,990

Notes:

1. Includes the Decontamination Facility, Technical Services Building, Gaseous Effluent Vent System Throughout Plant, Blending and Sampling, and Centrifuge Test and Post Mortem Facilities.
2. Supervision at 20%.
3. Supply ongoing monitoring and analysis service for dismantling teams.
4. Specific details of decontamination method not defined at this time.

Table 10.1-4 Restoration of Contaminated Areas on Facility Grounds (Work Days)

Activity	Labor Category	Labor Category	Labor Category	Labor Category	Labor Category	Labor Category
Backfill and Restore Site (Note 1)						
TOTAL						

Note:

1. Deviates from NUREG-1757 because cost is based on volume and unit cost associated with removal and disposal of liners and earthen covers of the facility Treated Effluent Evaporative Basin. The cost (see Table 10.1-14) assumes transport and disposal of approximately 33,000 ft^3 of contaminated soil and basin membrane. The cost of removal of the facility Treated Effluent Evaporative Basin material (33,000 ft^3) is based on a \$30/ft^3 disposal cost and includes the cost of excavation (\$5.00/yd^3 which includes labor and equipment costs) and cost of transportation (\$4.00/mile for approximately 1,100 miles from the NEF site to the Envirocare facility in Utah). Based on Urenco experience, other areas outside of the plant buildings are not expected to be contaminated.

Table 10.1-5 Final Radiation Survey
Page 1 of 1

Activity	Costs ($000)	Labor Shift-worker (multi-functional) (Man-days)	Labor Project Management (Man-days)	Labor HP&S (Man-days)	Activity Duration (Months)
Prepare Survey Plans and Grid Areas	500	439	334	360	8
Collect Survey Readings and Analyze Data	1,400 (Note 1)	1,261	343	1,013	16
Sample Analysis			568		
Final Status Survey Report and NRC Review	300	0	533	0	8
Confirmatory Survey and Report	200	0	355	0	6
Terminate Site License	100	0	178	0	2
TOTAL	2,500	1,700	2,311	1,373	(Note 2)

Notes:

1. The $1.4 million cost assigned to the conduct of the final radiation survey includes a cost of $365,000 to conduct the sampling and perform the sample analysis by a contractor. The sampling labor cost component ($45,000) was estimated assuming $60/hr (HP&S man-hour rate) for an estimated 500 samples with an average sample duration of 1.5 hours/sample. The analysis cost component ($320,000) for the 500 samples was estimated using a conservative $640/sample based on recent actual 2004 lab analysis costs. Because of the modeling for this activity, this sample analysis cost is expressed in terms of equivalent man-hours at the Project Management man-hour rate.
2. Some activities will be conducted in parallel to achieve a 36 month time frame.

Table 10.1-6 Site Stabilization and Long-Term Surveillance
(Work Days)
Page 1 of 1

Activity	Labor Category	Labor Category	Labor Category	Labor Category	Labor Category	Labor Category
(Note 1)	N/A	N/A	N/A	N/A	N/A	N/A

Note:

1. Urenco experience with decommissioning gas centrifuge uranium enrichment plants has been that there is no resultant ground contamination. As a result, site stabilization and long-term surveillance will not be required and associated decommissioning provisions are not provided.

Table 10.1-7 Total Work Days by Labor Category
(Based on a 7.5 hr Working Day)
Page 1 of 1

Task	Shift- worker (multi-functional)	Craftsman	Supervision	Project Management	HP&S	Cleaner
Planning and Preparation (see Table 10.1-2)	82	0	0	1,969	144	0
Decontamination and/or Dismantling of Radioactive Facility Components (Note 2)	56,067	1,896	6,156	1,478	1,828	2,897
Restoration of Contaminated Areas on Facility Grounds (Note 1) (see Table 10.1-4)	-	-	-	-	-	-
Final Radiation Survey (see Table 10.1-5)	1,700	0	0	2,311	1,373	0
Site Stabilization and Long-Term Surveillance (see Table 10.1-6)	0	0	0	0	0	0

Notes:

1. Cost estimate is activity-based.
2. The values shown are inclusive of the Separations Module input derived using the total costs in Table 10.1-9 and dividing by the cost per day for each labor category.

Table 10.1-8 Worker Unit Cost Schedule
Page 1 of 1

Labor Cost Component	Shift- worker (multi-functional)	Craftsman	Supervision	Project Management	HP&S	Cleaner
Salary & Fringe ($/year)	73,006	65,184	96,000	120,000	96,000	73,006
Overhead Rate (%)	excluded	excluded	excluded	excluded	excluded	excluded
Total Cost Per Year ($)	73,006	65,184	96,000	120,000	96,000	73,006
Total Cost Per Work Day ($/day) (Note 1)	342	306	450	563	450	342

Note:

1. Based on 213.33 work days per year at 7.5 hrs per day (1,600 hrs per year)

Table 10.1-9 Total Labor Costs by Major Decommissioning Task
($000)
Page 1 of 1

Task	Shift-worker (multi-functional)	Craftsman	Supervision	Project Management	HP&S	Cleaner
Planning and Preparation (see Table 10.1-2)	28	0	0	1,109	65	0
Decontamination and/or Dismantling of Radioactive Facility Components	19,175	579	2,770	832	823	991
Restoration of Contaminated Areas on Facility Grounds (Note 1) (see Table 10.1-4)	-	-	-	-	-	-
Final Radiation Survey (see Table 10.1-5)	581	0	0	1,301	618	0
Site Stabilization and Long-Term Surveillance (see Table 10.1-6)	0	0	0	0	0	0

Note:

1. Cost estimate is activity-based.

Table 10.1-10 Packaging, Shipping and Disposal of Radioactive Wastes
(Excluding Labor Costs)

(a) Waste Disposal Costs (includes packaging & shipping costs)

Waste Type	Disposal Volume (m^3 (ft^3))	Unit Cost ($/$ft^3$)	# of drums	Total Disposal Costs ($000)
Other Buildings :				
Miscellaneous low level waste	83 (2.930)	150	400	440
Separation Modules:				
Solidified Liquid Wastes	432 (15,251)	100	2,159	1,525
Centrifuge Components, Piping and Other Parts	1,036 (36,595)	100	5,180	3,659
Aluminum	3,602 (127,200)	100	NA	12,720
TOTAL	5,153 (181,976)	--	7,739	18,344

(b) Processing Costs

Materials	Disposal Weight (tons)	Unit Cost ($/lb)	Total Disposal Costs ($000)
Aluminum	10,177	0.14	2,860
Other materials	155	2.67	830
TOTAL	10,332	--	3,690

Table 10.1-11 Equipment and Supply Costs
(Excluded Containers)
Page 1 of 1

(a) Equipment

Equipment	Quantity	Unit Cost ($/unit)	Total Cost Equipment ($000)
Separation Building Modules			
Dismantling and decontamination building	45,210 ft^2	1,545	6,490
Special floor and vent system	45,210 ft^2	294	1,240
Plant equipment			
Basic decontamination equipment	lot (Note 1)	600,000	600
Decontamination line equipment	2 units	3,908,850	7,820
Evaporation installation	lot (Note 1)	390,000	390
Radiation and control equipment	lot (Note 1)	410,000	410
Electrical and Instrumentation			
Electrical system	lot (Note 1)	500,000	500
Instrumentation	lot (Note 1)	590,000	590
Design and Engineering			
Building	-	20% (Note 1)	1,550
Plant and equipment	-	15% (Note 1)	1,400
Electrical and Instrumentation	-	25% (Note 1)	270
Other Buildings:			
Dismantling/Cleaning Tools, Equipment and Consumables	lot (Note 1)	100,000	100
TOTAL	--	--	21,360

Note:
1. Allocation based on Urenco decommissioning experience.

(b) Supply

Equipment	Quantity	Unit Cost ($/$ft^3$)	Total Cost Equipment ($000)
Electricity kwh	2,910,344	0.062	180
Gas ft^3	16,900,000	0.004	75
Water ft^3	86,300	0.035	3
Materials	lot (Note 1)		653
TOTAL	--	--	910

Note:
1. Allocation based on Urenco decommissioning experience.

Table 10.1-12 Laboratory Costs
Page 1 of 1

Activity	Quantity	Unit Cost ($)	Total Costs ($000)
Analysis of samples	931	934	870
TOTAL	--	--	870

Table 10.1-13 Period Dependent Costs
Page 1 of 1

Cost Item	Total Cost ($000)
License Fees	(Note 1)
Insurance	(Note 1)
Taxes	(Note 1)
Other	(Note 1)
TOTAL	10,000

Note:

1. Period Dependent Costs include management, insurance, taxes, and other costs for the period beginning with the termination of operations of Separations Building Module 3 and the remaining plant facilities. This assumes $2,000,000 per year for each of the five years at the end of the project. It has been assumed that the period dependent decommissioning costs incurred during concurrent enrichment operations will be funded from operating plant funding and not the decommissioning trust fund.

Table 10.1-14 Total Decommissioning Costs
Page 1 of 2

(Note 7)

Task/Components	Costs ($000)		Total ($000)	Percentage	Notes
	Separations Modules	Other Buildings			
Planning and Preparation (see Table 10.1-2)	1,200	0	1,200	1%	1
Decontamination and Dismantling of Radioactive Facility Components (see Table 10.1-9)	24,060	1,110	25,170	20%	8
Restoration of Contamination Areas on Facility Grounds (see Table 10.1-4)	1,357	0	1,357	1%	2
Final Radiation Survey (see Table 10.1-5)	2,500	0	2,500	2%	3
Cost of Third Party Use	39,829	1,232	41,061	32%	11
Site Stabilization and Long-term Surveillance	0	0	0	0%	4
Waste Processing Costs (see Table 10.1-10)	3,690	0	3,690	3%	5
Waste Disposal Costs (see Table 10.1-10)	17,904	440	18,344	14%	6
Equipment Costs (see Table 10.1-11)	21,260	100	21,360	17%	--
Supply Costs (see Table 10.1-11)	910	0	910	1%	--
Laboratory Costs (see Table 10.1-12)	870	0	870	1%	--
Period Dependent Costs (see Table 10.1-13)	10,000	0	10,000	8%	--
SUBTOTAL (2002)	123,580	2,882	126,462		--
SUBTOTAL (with escalation to 2004)	128,115	2,988	131,103		12
Tails Disposition (2004)	--	--	622,169		9
Contingency (25%)	--	--	188,318		--
TOTAL (2004)	--	--	941,590		10

Table 10.1-14 Total Decommissioning Costs
Page 2 of 2

Notes:

1. The $1,200 includes planning, site characterization, Decommissioning Plan preparation, and NRC review for the entire plant.
2. Cost provided is for removal and disposal of liners and earthen covers of the facility Treated Effluent Evaporative Basin. The cost assumes transport and disposal of approximately 33,000 ft^3 of contaminated soil and basin membrane at recent commercial rates. The cost of removal of the facility Treated Effluent Evaporative Basin material (33,000 ft^3) is based on a $30/$ft^3$ disposal cost and includes the cost of excavation ($5.00/$yd^3$ which includes labor and equipment costs) and cost of transportation ($4.00/mile for approximately 1,100 miles from the NEF site to the Envirocare facility in Utah). Other areas outside of the plant buildings are not expected to be contaminated.
3. The $2,500 includes the Final Radiation Survey, NRC review, confirmatory surveys and license termination for the entire plant.
4. Site stabilization and long-term surveillance will not be required.
5. Waste processing costs are based on commercial metal melting equipment and unit rates obtained from Urenco experience in Europe.
6. Includes waste packaging and shipping costs. Waste disposal costs for Other Buildings are based on a $150 per cubic foot unit rate which includes packaging, shipping and disposal at Envirocare in Utah.
7. More than 97% of the decommissioning costs for the facility are attributed to the dismantling, decontamination, processing, and disposal of centrifuges and other equipment in the Separations Building Modules, which are considered classified. Given the classified nature of these buildings, the data presented in these Tables have been structured to meet the applicable NUREG-1757 recommendations, to the extent practicable. However, specific information such as numbers of components and unit rates has been intentionally excluded to protect the classified nature of the data. The remaining 3% of the decommissioning costs are for the remaining systems and components in Other Buildings.
8. The $1,110 for Other Buildings includes the decontamination and dismantling of contaminated equipment in the TBS, Blending and Liquid Sampling Area, Centrifuge Test and Post Mortem Facilities, and Gaseous Effluent Vent System.
9. Refer to Section 10.3, for Tails Disposition discussion.
10. Combined total for both decommissioning and tails disposition.
11. An adjustment has been applied to account for use of a third party for performing decommissioning operations associated with planning and preparation, decontamination and dismantling of radioactive facility components, restoration of contaminated grounds, and the final radiation survey. The adjustment includes an overhead rate on direct staff labor of 110%, plus 15% profit on labor and its overheads.
12. The escalation cost factor applied is based on the Gross Domestic Product (GDP) implicit price deflator. The resulting escalation cost factor for January 2002 to January 2004 is a 3.67% increase. The escalation cost factor is not applied to the tails disposition costs since these costs are provided in 2004 dollars.

Table 10.3-1 Summary of Depleted UF_6 Disposal Costs from Four Sources
Page 1 of 1

Source	Costs in 2002 Dollars per kgU			
	Conversion	Disposal	Transportation	Total
LLNL (UCRL-AR-127650) (a)	2.64	2.17	0.25	5.06
UDS Contract (b)	(d)	(d)	(d)	3.92
URENCO (e)	(d)	(d)	(d)	(d)
CEC Cost Estimate (c)	4.93	1.47	0.34	6.74

Notes:

(a) 1997 Lawrence Livermore National Laboratory cost estimate study for DOE, discounted costs in 1996 dollars were undiscounted and escalated to 2002 by ERI.
(b) Uranium Disposition Services (UDS) contract with DOE for capital and operating costs for first five years of Depleted UF_6 conversion and Depleted U_3O_8 conversion product disposition.
(c) Based upon Depleted UF_6 and Depleted U_3O_8 disposition costs provided to the NRC during Claiborne Enrichment Center license application in 1993.
(d) Cost component is proprietary or not made available.
(e) The average of the three costs is \$5.24/kg U. LES has selected \$5.50/kg U as the disposal cost for the National Enrichment Facility. Urenco has reviewed this cost estimate, and based on its current experience with UF_6 disposal, finds this figure to be prudent.

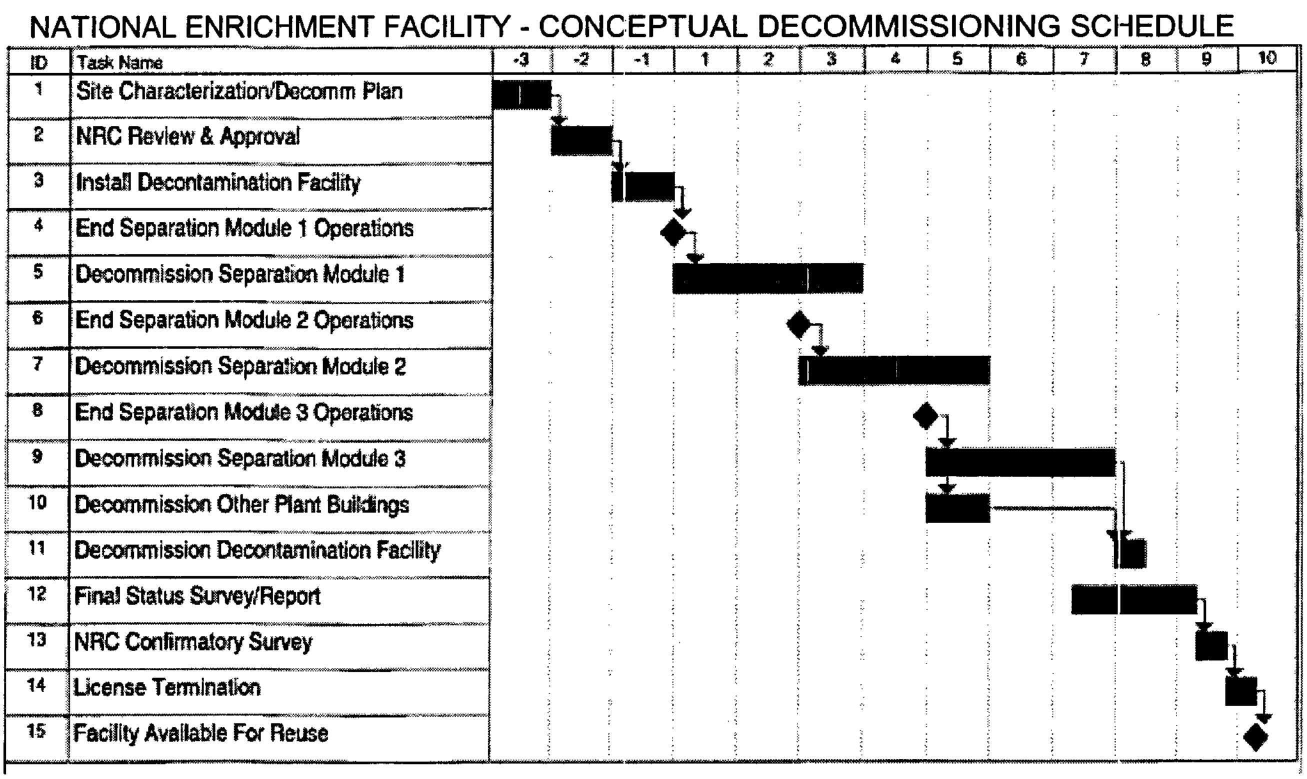
Table 10.3-2 DOE-UDS August 29, 2002, Contract Quantities and Costs
Page 1 of 1

	Target Million kgU	
UDS Conversion and Disposal Quantities:	DUF6 (a)	U (b)
FY 2005 (August-September)	1.050	0.710
FY 2006	27.825	18.800
FY 2007	31.500	21.294
FY 2008	31.500	21.294
FY 2009	31.500	21.294
FY 2010 (October-July)	26.250	17.745
Total:	149.625	101.147
Nominal Conversion Rate (c) and Target Conversion Rate (Million kgU/Yr)		21.3
UDS Contract Workscope Costs: (d)		**Million $**
Design, Permitting, Project Management, etc.		27.99
Construct Paducah Conversion Facility		93.96
Construct Portsmouth Conversion Facility		90.40
Operations for First 5 Years DUF_6 and DU_3O_8 (e)		283.23
Contract Estimated Total Cost w/o Fee		495.58
Contract Estimated Value per DOE PR, August 29, 2003		558.00
Difference Between Cost and Value is the Estimated Fee of 12.6%		62.42
Capital Cost w/o Fee		212.35
Capital Cost with Fee		239.10
First 5 Years Operating Cost with Fee		318.92
Estimated Unit Conversion and Disposal Costs:		
Unit Capital Cost (f)		$0.77/kgU
2005-2010 Unit Operating Costs in 2002 $		$3.15/kgU
Total Estimated Unit Cost		$3.92/kgU

Notes:

(a) As on page B-10 of the UDS contract.
(b) DUF_6 weight multiplied by the uranium atomic mass fraction, 0.676.
(c) Based on page H-34 of the UDS contract.
(d) Workscope costs as on UDS contract pages B-2 and B-3.
(e) Does not include any potential off-set credit for HF sales.
(f) Assumed operation over 25 years, 6% government cost of money, and no taxes.

FIGURES



REFERENCE NUMBER
Figure 10.1-1.doc

FIGURE 10.1-1

NATIONAL ENRICHMENT FACILITY – CONCEPTUAL DECOMMISSIONING SCHEDULE

REVISION DATE: DECEMBER 2003

APPENDIX 10A

PAYMENT SURETY BOND

Date bond executed: ____________

Effective date: ____________

Principal: Louisiana Energy Services, L.P.
100 Sun Avenue NE, Suite 204
Albuquerque, NM 87109

Type of organization: Limited Partnership

State of incorporation: Delaware

NRC license number, name and address of facility, and amount for decommissioning activities guaranteed by this bond: ____________

Surety: [*Insert name and business address*]

Type of organization: [*Insert "proprietorship," "partnership," or "corporation"*]

State of incorporation: ____________ *(if applicable)*

Surety's qualification in jurisdiction where licensed facility is located.

Surety's bond number: ____________

Total penal sum of bond: $____________

Know all persons by these presents, that we, the Principal and Surety hereto, are firmly bound to the U.S. Nuclear Regulatory Commission (hereinafter called NRC) in the above penal sum for the payment of which we bind ourselves, our heirs, executors, administrators, successors, and assigns jointly and severally; provided that, where the Sureties are corporations acting as co-sureties, we, the Sureties, bind ourselves in such sum "jointly and severally" only for the purpose of allowing a joint action or actions against any or all of us, and for all other purposes each Surety binds itself, jointly and severally with the Principal, for the payment of such sum only as is set forth opposite the name of such Surety; but if no limit of liability is indicated, the limit of liability shall be the full amount of the penal sum.

WHEREAS, the NRC, an agency of the U.S. Government, pursuant to the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974, has promulgated regulations in title 10, Chapter I of the *Code of Federal Regulations*, Parts 30, 40, and 70, applicable to the Principal, which require that a license holder or an applicant for a facility license provide financial assurance that funds will be available when needed for facility decommissioning;

NOW, THEREFORE, the conditions of the obligation are such that if the Principal shall faithfully, before the beginning of decommissioning of each facility identified above, fund the standby trust fund in the amount(s) identified above for the facility;

Or, if the Principal shall fund the standby trust fund in such amount(s) after an order to begin facility decommissioning is issued by NRC or a U.S. District Court or other court of competent jurisdiction;

Or, if the Principal shall provide alternative financial assurance, and obtain NRC's written approval of such assurance, within 30 days after the date a notice of cancellation from the Surety is received by both the Principal and NRC, then this obligation shall be null and void; otherwise it is to remain in full force and effect.

The Surety shall become liable on this bond obligation only when the Principal has failed to fulfill the conditions described above. Upon notification by NRC that the Principal has failed to perform as guaranteed by this bond, the Surety shall place funds in the amount guaranteed for the facility into the standby trust fund.

The liability of the Surety shall not be discharged by any payment or succession of payments hereunder, unless and until such payment or payments shall amount in the aggregate to the penal sum of the bond, but in no event shall the obligation of the Surety hereunder exceed the amount of said penal sum.

The Surety may cancel the bond by sending notice of cancellation by certified mail to the Principal and to NRC provided, however, that cancellation shall not occur during the 90 days beginning on the date of receipt of the notice of cancellation by both the Principal and NRC, as evidenced by the return receipts.

The Principal may terminate this bond by sending written notice to NRC and to the Surety 90 days prior to the proposed date of termination, provided, however, that no such notice shall become effective until the Surety receives written authorization for termination of the bond from NRC.

The Principal and Surety hereby agree to adjust the penal sum of the bond yearly so that it guarantees a new amount, provided that the penal sum does not increase by more than 20 percent in any one year and no decrease in the penal sum takes place without the written permission of NRC.

If any part of this agreement is invalid, it shall not affect the remaining provisions that will remain valid and enforceable.

In Witness Whereof, the Principal and Surety have executed this financial guarantee bond and have affixed their seals on the date set forth above.

The persons whose signatures appear below hereby certify that they are authorized to execute this surety bond on behalf of the Principal and Surety.

Principal

[*Signatures*]
E. James Ferland
President, Louisiana Energy Services, L.P.
[*Corporate seal*]

Corporate Surety

[*Name and address*]

State of incorporation: ___________

Liability limit: $___________

[*Signatures*]
[*Names and titles*]
[*Corporate seal*]

Bond Premium: $___________

APPENDIX 10B

STANDBY TRUST AGREEMENT

TRUST AGREEMENT, the Agreement entered into as of [*insert date*] by and between Louisiana Energy Service, L. P., a Delaware limited partnership, herein referred to as the "Grantor," and [*insert name and address of a trustee acceptable to NRC*], the "Trustee."

WHEREAS, the U.S. Nuclear Regulatory Commission (NRC), an agency of the U.S.

Government, pursuant to the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974, has promulgated regulations in title 10, Chapter I, of the *Code of Federal Regulations*, Parts 30, 40, and 70. These regulations, applicable to the Grantor, require that a holder of, or an applicant for, a materials license issued pursuant to 10 CFR Parts 30, 40, and 70 provide assurance that funds will be available when needed for required decommissioning activities.

WHEREAS, the Grantor has elected to use a surety bond to provide all of such financial assurance for the facilities identified herein; and

WHEREAS, when payment is made under a surety bond, this standby trust shall be used for the receipt of such payment; and

WHEREAS, the Grantor, acting through its duly authorized officers, has selected the Trustee to be the trustee under this Agreement, and the Trustee is willing to act as trustee;

NOW, THEREFORE, the Grantor and the Trustee agree as follows:

<u>Section 1. Definitions.</u> As used in this Agreement:

(a)The term "Grantor" means the NRC licensee who enters into this Agreement and any successors or assigns of the Grantor.

(b) The term "Trustee" means the trustee who enters into this Agreement and any successor trustee.

<u>Section 2. Costs of Decommissioning.</u> This Agreement pertains to the costs of decommissioning the materials and activities identified in License Number [*insert license number*] issued pursuant to 10 CFR Parts 30, 40, and 70, as shown in Schedule A.

<u>Section 3. Establishment of Fund.</u> The Grantor and the Trustee hereby establish a standby trust fund (the Fund) for the benefit of NRC. The Grantor and the Trustee intend that no third party shall have access to the Fund except as provided herein.

<u>Section 4. Payments Constituting the Fund.</u> Payments made to the Trustee for the Fund shall consist of cash, securities, or other liquid assets acceptable to the Trustee. The Fund is established initially as consisting of the property, which is acceptable to the Trustee , described

in Schedule B attached hereto. Such property and any other property subsequently transferred to the Trustee are referred to as the "Fund," together with all earnings and profits thereon, less any payments or distributions made by the Trustee pursuant to this Agreement. The Fund shall be held by the Trustee, IN TRUST, as hereinafter provided. The Trustee shall not be responsible nor shall it undertake any responsibility for the amount of, or adequacy of the Fund, nor any duty to collect from the Grantor, any payments necessary to discharge any liabilities of the Grantor established by NRC.

Section 5. Payment for Required Activities Specified in the Plan. The Trustee shall make payments from the Fund to the Grantor upon presentation to the Trustee of the following:

(a) A certificate duly executed by the Secretary of the Grantor's Management Committee attesting to the occurrence of the events, and in the form set forth in the attached Certificate of Events, and

(b) A certificate attesting to the following conditions:

  (1) that decommissioning is proceeding pursuant to an NRC-approved plan;

  (2) that the funds withdrawn will be expended for activities undertaken pursuant to that plan; and

  (3) that NRC has been given 30 days prior notice of Louisiana Energy Service's intent to withdraw funds from the trust fund.

No withdrawal from the Fund for a particular license can exceed 10 percent of the remaining funds available for that license unless NRC written approval is attached.

In addition, the Trustee shall make payments from the Fund as NRC shall direct, in writing, to provide for the payment of the costs of required activities covered by this Agreement. The Trustee shall reimburse the Grantor or other persons as specified by NRC from the Fund for expenditures for required activities in such amounts as NRC shall direct in writing. In addition, the Trustee shall refund to the Grantor such amounts as NRC specifies in writing. Upon refund, such funds shall no longer constitute part of the Fund as defined herein.

Section 6. Trust Management. The Trustee shall invest and reinvest the principal and income of the Fund and keep the Fund invested as a single fund, without distinction between principal and income, in accordance with general investment policies and guidelines which the Grantor may communicate in writing to the Trustee from time to time, subject, however, to the provisions of this section. In investing, reinvesting, exchanging, selling, and managing the Fund, the Trustee shall discharge its duties with respect to the Fund solely in the interest of the beneficiary and with the care, skill, prudence and diligence under the circumstances then prevailing which persons of

prudence, acting in a like capacity and familiar with such matters, would use in the conduct of an enterprise of a like character and with like aims, except that:

(a) Securities or other obligations of the Grantor, or any other owner or operator of the facilities, or any of their affiliates as defined in the Investment Company Act of 1940, as amended (15 U.S.C. 80a-2(a)), shall not be acquired or held, unless they are securities or other obligations of the Federal or a State government;

(b) The Trustee is authorized to invest the Fund in time or demand deposits of the Trustee, to the extent insured by an agency of the Federal government, and in obligations of the Federal government such as GNMA, FNMA, and FHLM bonds and certificates or State and Municipal bonds rated BBB or higher by Standard & Poor's or Baa or higher by Moody's Investment Services; and

(c) For a reasonable time, not to exceed 60 days, the Trustee is authorized to hold uninvested cash, awaiting investment or distribution, without liability for the payment of interest thereon.

Section 7. Commingling and Investment. The Trustee is expressly authorized in its discretion:

(a) To transfer from time to time any or all of the assets of the Fund to any common, commingled, or collective trust fund created by the Trustee in which the Fund is eligible to participate, subject to all of the provisions thereof, to be commingled with the assets of other trusts participating therein; and

(b) To purchase shares in any investment company registered under the Investment Company Act of 1940 (15 U.S.C. 80a-1 et seq.), including one that may be created, managed, underwritten, or to which investment advice is rendered, or the shares of which are sold by the Trustee. The Trustee may vote such shares in its discretion.

Section 8. Express Powers of Trustee. Without in any way limiting the powers and discretion conferred upon the Trustee by the other provisions of this Agreement or by law, the Trustee is expressly authorized and empowered:

(a) To sell, exchange, convey, transfer, or otherwise dispose of any property held by it, by public or private sale, as necessary to allow duly authorized withdrawals at the joint request of the Grantor and NRC or to reinvest in securities at the direction of the Grantor;

(b) To make, execute, acknowledge, and deliver any and all documents of transfer and conveyance and any and all other instruments that may be necessary or appropriate to carry out the powers herein granted;

(c) To register any securities held in the Fund in its own name, or in the name of a nominee, and to hold any security in bearer form or in book entry, or to combine certificates representing such securities with certificates of the same issue held by the Trustee in other fiduciary capacities, to reinvest interest payments and funds from matured and redeemed instruments, to file proper forms concerning securities held in the Fund in a timely fashion with appropriate government agencies, or to deposit or arrange for the deposit of such securities in a qualified central depository even though, when so deposited, such securities may be merged and held in bulk in the name of the nominee or such depository with other securities deposited therein by another person, or to deposit or arrange for the deposit of any securities issued by the U.S. Government, or any agency or instrumentality thereof, with a Federal Reserve Bank, but the books and records of the Trustee shall at all times show that all such securities are part of the Fund;

(d) To deposit any cash in the Fund in interest-bearing accounts maintained or savings certificates issued by the Trustee, in its separate corporate capacity, or in any other banking institution affiliated with the Trustee, to the extent insured by an agency of the Federal government; and

(e) To compromise or otherwise adjust all claims in favor of or against the Fund.

Section 9. Taxes and Expenses. All taxes of any kind that may be assessed or levied against or in respect of the Fund and all brokerage commissions incurred by the Fund shall be paid from the Fund. All other expenses incurred by the Trustee in connection with the administration of this Trust, including fees for legal services rendered to the Trustee, the compensation of the Trustee to the extent not paid directly by the Grantor, and all other proper charges and disbursements of the Trustee shall be paid from the Fund.

Section 10. Annual Valuation. After payment has been made into this standby trust fund, the Trustee shall annually, at least 30 days before the anniversary date of receipt of payment into the standby trust fund, furnish to the Grantor and to NRC a statement confirming the value of the Trust. Any securities in the Fund shall be valued at market value as of no more than 60 days before the anniversary date of the establishment of the Fund. The failure of the Grantor to object in writing to the Trustee within 90 days after the statement has been furnished to the Grantor and NRC shall constitute a conclusively binding assent by the Grantor, barring the Grantor from asserting any claim or liability against the Trustee with respect to the matters disclosed in the statement.

Section 11. Advice of Counsel. The Trustee may from time to time consult with counsel with respect to any question arising as to the construction of this Agreement or any action to be taken hereunder. The Trustee shall be fully protected, to the extent permitted by law, in acting on the advice of counsel.

Section 12. Trustee Compensation. The Trustee shall be entitled to reasonable compensation for its services as agreed upon in writing with the Grantor. (See Schedule C.)

Section 13. Successor Trustee. Upon 90 days notice to NRC and the Grantor, the Trustee may resign; upon 90 days notice to NRC and the Trustee, the Grantor may replace the Trustee; but such resignation or replacement shall not be effective until the Grantor has appointed a successor Trustee, the successor accepts the appointment, the successor is ready to assume its duties as trustee, and NRC has agreed, in writing, that the successor is an appropriate Federal or State government agency or an entity that has the authority to act as a trustee and whose trust operations are regulated and examined by a Federal or State agency. The successor Trustee shall have the same powers and duties as those conferred upon the Trustee hereunder. When the resignation or replacement is effective, the Trustee shall assign, transfer, and pay over to the successor Trustee the funds and properties then constituting the Fund. If for any reason the Grantor cannot or does not act in the event of the resignation of the Trustee, the Trustee may apply to a court of competent jurisdiction for the appointment of a successor Trustee or for instructions. The successor Trustee shall specify the date on which it assumes administration of the trust, in a writing sent to the Grantor, NRC, and the present Trustee, by certified mail 10 days before such change becomes effective. Any expenses incurred by the Trustee as a result of any of the acts contemplated by this section shall be paid as provided in Section 9.

Section 14. Instructions to the Trustee. All orders, requests, and instructions by the Grantor to the Trustee shall be in writing, signed by such persons as are signatories to this Agreement or such other designees as the Grantor may designate in writing. The Trustee shall be fully protected in acting without inquiry in accordance with the Grantor's orders, requests, and instructions. If NRC issues orders, requests, or instructions to the Trustee these shall be in writing, signed by NRC or its designees, and the Trustee shall act and shall be fully protected in acting in accordance with such orders, requests, and instructions. The Trustee shall have the right to assume, in the absence of written notice to the contrary, that no event constituting a change or a termination of the authority of any person to act on behalf of the Grantor or NRC hereunder has occurred. The Trustee shall have no duty to act in the absence of such orders, requests, and instructions from the Grantor and/or NRC, except as provided for herein.

Section 15. Amendment of Agreement. This Agreement may be amended by an instrument in writing executed by the Grantor, the Trustee, and NRC, or by the Trustee and NRC if the Grantor ceases to exist. All amendments shall meet the relevant regulatory requirements of NRC.

Section 16. Irrevocability and Termination. Subject to the right of the parties to amend this Agreement as provided in Section 15, this trust shall be irrevocable and shall continue until terminated at the written agreement of the Grantor, the Trustee, and NRC, or by the Trustee and NRC if the Grantor ceases to exist. Upon termination of the trust, all remaining trust property, less final trust administration expenses, shall be delivered to the Grantor or its successor.

Section 17. Immunity and Indemnification. The Trustee shall not incur personal liability of any nature in connection with any act or omission, made in good faith, in the administration of this trust, or in carrying out any directions by the Grantor or NRC issued in accordance with this Agreement. The Trustee shall be indemnified and saved harmless by the Grantor or from the trust fund, or both, from and against any personal liability to which the Trustee may be subjected by reason of any act or conduct in its official capacity, including all expenses reasonably incurred in its defense in the event the Grantor fails to provide such defense.

Section 18. This Agreement shall be administered, construed, and enforced according to the laws of the State of [*insert name of State*].

Section 19. Interpretation and Severability. As used in this Agreement, words in the singular include the plural and words in the plural include the singular. The descriptive headings for each section of this Agreement shall not affect the interpretation or the legal efficacy of this Agreement. If any part of this Agreement is invalid, it shall not affect the remaining provisions which will remain valid and enforceable.

IN WITNESS WHEREOF the parties have caused this Agreement to be executed by the respective officers duly authorized and the incorporate seals to be hereunto affixed and attested as of the date first written above.

Louisiana Energy Services, L. P.
[*Signature of E. James Ferland*]
E. James Ferland
President, Louisiana Energy Services, L. P

ATTEST:
[*Title*]
[*Seal*]

[*Insert name and address of Trustee*]
[*Signature of representative of Trustee*]
[*Title*]

ATTEST:
[*Title*]
[*Seal*]

APPENDIX 10C

STANDBY TRUST AGREEMENT SCHEDULES

Schedule A

This Agreement demonstrates financial assurance for the following cost estimates or prescribed amounts for the following licensed activities:

U.S. NUCLEAR REGULATORY COMMISSION LICENSE NUMBER(S)	NAME AND ADDRESS OF LICENSEE	ADDRESS OF LICENSED ACTIVITY	COST ESTIMATES FOR REGULATORY ASSURANCES DEMONSTRATED BY THIS AGREEMENT
	Louisiana Energy Services, L.P. 100 Sun Avenue NE, Suite 204 Albuquerque, NM 87109		

The cost estimates listed here were last adjusted and approved by NRC on [*insert date*].

Schedule B

DOLLAR AMOUNT________________

AS EVIDENCED BY________________

Schedule C

[*Insert name, address, and phone number of Trustee.*]
Trustee's fees shall be $________________per year.

APPENDIX D
SPECIMEN CERTIFICATE OF EVENTS

[*Insert name and address of trustee*]

Attention: Trust Division

Gentlemen:

In accordance with the terms of the Agreement with you dated _______, I, ________, Secretary of the Management Committee of Louisiana Energy Services, L. P., hereby certify that the following events have occurred:

1. Louisiana Energy Services, L. P., is required to commence the decommissioning of its facility located in Lea County, New Mexico (hereinafter called the decommissioning).

2. The plans and procedures for the commencement and conduct of the decommissioning have been approved by the United States Nuclear Regulatory Commission, or its successor, on ______________(copy of approval attached).

3. The Management Committee of Louisiana Energy Services, L. P., has adopted the attached resolution authorizing the commencement of the decommissioning.

____

Secretary of the Management Committee of
Louisiana Energy Services, L. P.

____

Date

APPENDIX 10E

SPECIMEN CERTIFICATE OF RESOLUTION

I, ________, do hereby certify that I am Secretary of the Management Committee of Louisiana Energy Services, L. P., a Delaware Limited Partnership, and that the resolution listed below was duly adopted at a meeting of this Limited Partnership's Management Committee on _________, 20__.

IN WITNESS WHEREOF, I have hereunto signed my name and affixed the seal of this Limited Partnership this ___ day of _________, 20__.

____

Secretary of the Management Committee of
Louisiana Energy Services, L. P.

RESOLVED, that this Management Committee hereby authorizes the President, or such other employee of the Limited Partnership as he may designate, to commence decommissioning activities at the National Enrichment Facility in accordance with the terms and conditions described to this Management Committee at this meeting and with such other terms and conditions as the President shall approve with and upon the advice of Counsel.

APPENDIX 10F
LETTER OF ACKNOWLEDGMENT

STATE OF ________________

To Wit: ________________

CITY OF ________________

On this ___ day of ___________, before me, a notary public in and for the city and State aforesaid, personally appeared ___________, and she/he did depose and say that she/he is the [*insert title*] of ________________[*if applicable, insert ", national banking association" or ", State banking association"*], Trustee, which executed the above instrument; that she/he knows the seal of said association; that the seal affixed to such instrument is such corporate seal; that it was so affixed by order of the association; and that she/he signed her/his name thereto by like order.

____
[*Signature of notary public*]

My Commission Expires: ___________
[*Date*]

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11.0 MANAGEMENT MEASURES

Management measures are functions applied to item(s) relied on for safety (IROFS) and any items which may affect the function of IROFS to provide reasonable assurance that the IROFS are available and able to perform their functions when needed. This chapter addresses each of the management measures included in the 10 CFR 70.4 definition of management measures.

Management measures are implemented through a quality assurance (QA) program in accordance with 10 CFR 50, Appendix B (CFR, 2003b). The QA program also provides additional measures for ensuring that the design, construction, operation and decommissioning of IROFS are controlled commensurate with their importance to safety. The Louisiana Energy Services (LES) Quality Assurance Program is described in the LES QA Program Description document included as Appendix A to this chapter. The NRC has evaluated the LES QA Program Description and concluded that the application of QA elements as described in the QA Program Description meets the requirements of 10 CFR 70 (CFR, 2003g) and provides reasonable assurance of protection of public and worker health and safety and the environment (NRC, 2004). The current LES QA Program is also consistent with the QA Program submitted for Nuclear Regulatory Commission (NRC) review in Chapter 10 of the Claiborne Enrichment Center Safety Analysis Report (LES, 1993). The NRC staff evaluated the previous LES QA Program and concluded that the program, when implemented effectively, will meet the requirements of 10 CFR 50, Appendix B (CFR, 1994). The staff concluded in Section 12.3 of NUREG-1491 (NRC, 1994) that the LES QA program was acceptable for the design, construction, start-up, and operation of the enrichment facility. References to the NUREG-1491 (NRC, 1994) sections that document the NRC staff's previous acceptance of these management measures are included in each section as appropriate.

LES maintains full responsibility for assuring that the National Enrichment Facility (NEF) is designed, constructed, tested, and operated in conformance with good engineering practices, applicable regulatory requirements and specified design requirements and in a manner to protect the health and safety of the public. To this end, the LES Quality Assurance Program conforms to the criteria established in 10 CFR 50, Appendix B, Quality Assurance Criteria For Nuclear Power Plants and Fuel Reprocessing Plants (CFR, 2003b). The criteria in 10 CFR 50, Appendix B (CFR, 2003b), are implemented following the commitment to ASME NQA-1-1994, Quality Assurance Program Requirements for Nuclear Facilities (ASME, 1994), as revised by the ASME NQA-1a-1995 Addenda (ASME, 1995).

The QA Program described herein includes design, construction, pre-operational testing, and operation of the facility. This QA Program describes the requirements to be applied for those systems, components, items, and services that have been determined to be QA Level 1 as defined in Appendix A. LES and their contractors implement these requirements through the use of approved procedures. In addition, a quality assurance program as described in Appendix A is applied to certain other systems, components, items, and services which are not QA Level 1. The information provided in this chapter, the corresponding regulatory requirement, and the section of NUREG-1520 (NRC, 2002), Chapter 11 in which the NRC acceptance criteria are presented is summarized below.

Information Category and Requirement	10 CFR 70 Citation	NUREG-1520 Chapter 11 Reference
Section 11.1 Configuration Management	70.62(d) & 70.72	11.4.3.1
Section 11.2 Maintenance	70.62(d)	11.4.3.2
Section 11.3 Training and Qualifications	70.62(d) & 10CFR19	11.4.3.3
Section 11.4 Procedures Development and Implementation	70.62(d) & 70.22(a)(8)	11.4.3.4
Section 11.5 Audits and Assessments	70.62(d)	11.4.3.5
Section 11.6 Incident Investigations and Corrective Action Process	70.74(a)&(b) 70.62(a)(3)	11.4.3.6
Section 11.7 Records Management	70.62(a)(2)&(3) 70.62(d)	11.4.3.7
Section 11.8 Other QA Elements	70.62(d)	11.4.3.8
Appendix A: LES QA Program Description	70.62(d)	11.4.3.8

11.1 CONFIGURATION MANAGEMENT (CM)

This section describes the configuration management program for the NEF. Configuration management for the NEF is implemented through requirements of the QA Program and associated procedures.

The LES President is the executive responsible for quality assurance and is the highest level of management responsible for LES's QA policies, goals, and objectives. The President receives policy direction from the LES Management Committee. The LES organization during the design, construction and operation phases, including QA, is presented in Chapter 2, Organization and Administration.

11.1.1 Configuration Management Policy

Configuration management is provided throughout facility design, construction, testing, and operation. Configuration management provides the means to establish and maintain a technical baseline for the facility based on clearly defined requirements. During design and construction, the Engineering and Contracts Manager has responsibility for configuration management through the design control process. Selected documentation, including the integrated safety analysis (ISA), is controlled under the configuration management system in accordance with procedures associated with design control, document control, and records management. Design changes undergo formal review, including interdisciplinary reviews as appropriate, in accordance with these procedures. This interdisciplinary review includes as a minimum the review for ISA impacts.

Configuration management provides the means to establish and maintain the essential features of the design basis of IROFS, including the ISA. As the project progresses from design and construction to operation, configuration management is maintained by the Technical Services organization as the overall focus of activities changes. Procedures will define the turnover process and responsibilities since construction will continue on new work modules during operations.

During the design phase of the project, configuration management is based on the design control provisions and associated procedural controls over design documents to establish and maintain the technical baseline. Design documents, including the ISA, that provide design input, design analysis, or design results specifically for IROFS are identified with the appropriate QA level. These design documents undergo interdisciplinary review during the initial issue and during each subsequent revision. During the construction phase of the project, changes to drawings and specifications issued for construction, procurement, or fabrication are systematically reviewed and verified, evaluated for impact, including impact to the ISA, and approved prior to implementation. Proper implementation is verified and reflected in the design basis documentation.

In order to provide for the continued safe and reliable operation of the facility structures, systems and components, measures are implemented to ensure that the quality of these structures, systems and components is not compromised by planned changes (modifications). After issuance of the Operating License, the Plant Manager is responsible for the design of and modifications to facility structures, systems or components. The design and implementation of modifications are performed in a manner so as to assure quality is maintained in a manner commensurate with the remainder of the system which is being modified, or as dictated by applicable regulations.

The administrative instructions for modifications during the operations phase are contained in procedures that are approved, including revisions, by the Technical Services Manager. The modification procedure contains the following items necessary to ensure quality in the modification program:

- The technical and quality requirements which shall be met to implement a modification
- The requirements for initiating, approving, monitoring, designing, verifying, and documenting modifications. The facility modification procedure shall be written to ensure that policies are formulated and maintained to satisfy the LES QA Program, as applicable.

Each change to the facility or to activities of personnel shall have an evaluation performed in accordance with the requirements of 10 CFR 70.72 (CFR, 2003e), as applicable. Each modification shall also be evaluated for any required changes or additions to the facility's procedures, personnel training, testing program, or regulatory documents.

For any change (i.e., new design or operation, or modification to the facility or to activities of personnel, e.g., site structures, systems, components, computer programs, processes, operating procedures, management measures), that involves or could affect uranium on site, a Nuclear Criticality Safety (NCS) evaluation and, if required, an NCS analysis shall be prepared and approved. Prior to implementing the change, it shall be determined that the entire process will be subcritical (with applicable margin for safety) under both normal and credible abnormal conditions.

Each modification is also evaluated and documented for radiation exposure to minimize worker exposures in keeping with the facility as low as reasonably achievable (ALARA) program, criticality and worker safety requirements and/or restrictions. Other areas of consideration in evaluating modifications may include, but are not limited to the review of:

- Modification cost
- Lessons learned from similar completed modifications
- QA requirements
- Potential operability or maintainability concerns
- Constructability concerns
- Post-modification testing requirements
- Environmental considerations
- Human factors
- Integrated safety analysis.

After completion of a modification to a structure, system, or component, the modification Project Manager, or designee, shall ensure that all applicable testing has been completed to ensure correct operation of the system(s) affected by the modification and documentation regarding the modification is complete. In order to ensure operators are able to operate a modified system safely, when a modification is complete, all documents necessary, e.g., the revised process description, checklists for operation and flowsheets are made available to operations and maintenance departments prior to the start-up of the modified system. Appropriate training on the modification is completed before a system is placed in operation. A formal notice of a modification being completed is distributed to all appropriate managers. As-built drawings incorporating the modification are completed in accordance with the design control procedures. These records shall be identifiable and shall be retained in accordance with the records management procedures.

11.1.1.1 Scope of Structures, Systems, and Components

The scope of Structures, Systems, and Components (SSC) under configuration management includes all IROFS identified by the integrated safety analysis of the design bases and any items which may affect the function of the IROFS. Design documents subject to configuration management include calculations, safety analyses, design criteria, engineering drawings, system descriptions, technical documents, and specifications that establish design requirements for IROFS. During the design phase, these design documents are maintained under configuration management when initially approved.

The scope of documents included in the configuration management program expands throughout the design process. As drawings and specification sections related to IROFS or items affecting the functions of IROFS are prepared and issued for procurement, fabrication, or construction, these documents are included in configuration management.

During construction, initial startup, and operations, the scope of documents under configuration management similarly expands to include, as appropriate: vendor data; test data; inspection data; initial startup, test, operating and administrative procedures as applicable to IROFS and nonconformance reports. These documents include documentation related to IROFS that is generated through functional interfaces with QA, maintenance, and training and qualifications of personnel. Configuration management procedures will provide for evaluation, implementation, and tracking of changes to IROFS, and processes, equipment, computer programs, and activities of personnel that impact IROFS.

11.1.1.2 Interfaces with Other Management Measures

Configuration management is implemented through or otherwise related to other management measures. Key interfaces and relationships to other management measures are described below:

- **Quality Assurance -** The QA program establishes the framework for configuration management and other management measures for IROFS and items that affect the function of the IROFS.

- **Records Management** - Records associated with IROFS and items affecting IROFS are generated and processed in accordance with the applicable requirements of the QA Program and provide evidence of the conduct of activities associated with the configuration management of those IROFS.

- **Maintenance** – Maintenance requirements are established as part of the design basis, which is controlled under configuration management. Maintenance records for IROFS and items affecting IROFS provide evidence of compliance with preventative and corrective maintenance schedules.

- **Training and Qualifications** - Training and qualification are controlled in accordance with the applicable provisions of the QA Program. Personnel qualifications and/or training to specific processes and procedures are management measures that support the safe operation, maintenance, or testing of IROFS. Also, work activities that are themselves IROFS, (i.e., administrative controls) are proceduralized, and personnel are trained and qualified to these procedures. Training and qualification requirements and documentation of training may be considered part of the design basis controlled under configuration management. Reference Sections 11.3.2, Analysis and Identification of Functional Areas Requiring Training, and 11.3.3, Position Training Requirements, for interfaces with configuration management.

- **Incident Investigation/Audits and Assessments** - Audits, assessments, and incident investigations are described in Sections 11.5, Audits and Assessments, and 11.6, Incident Investigations and Corrective Action Process. Corrective actions identified as a result of these management measures may result in changes to design features, administrative controls, or other management measures (e.g., operating procedures). The Corrective Action Program (CAP) is described in Section 11.6, "Incident Investigations and Corrective Action Process." Changes are evaluated under the provisions of configuration management through the QA Program and procedures. Periodic assessments of the configuration management program are also conducted in accordance with the audit and assessment program described in Section 11.5.

- **Procedures** - Operating, administrative, maintenance, and emergency procedures are used to conduct various operations associated with IROFS and items affecting IROFS and will be reviewed for potential impacts to the design basis. Also, work activities that are themselves IROFS, (i.e., administrative controls) are contained in procedures.

11.1.1.3 Objectives of Configuration Management

The objectives of configuration management are to ensure design and operation within the design basis of IROFS by: identifying and controlling preparation and review of documentation associated with IROFS; controlling changes to IROFS; and maintaining the physical configuration of the facility consistent with the approved design.

The Urenco technology transfer documentation provides the enrichment plant design, and identifies those safety trips and features credited in the European safety analyses. The ISA of the design bases determines the IROFS and establishes the safety function(s) associated with procedures for controlling design, including preparation, review (including interdisciplinary review), design verification where appropriate, approval, and release and distribution for use. Engineering documents will be assessed for QA level classification. Changes to the approved design are subject to a review to ensure consistency with the design bases of IROFS. Configuration verification is also accomplished through design verification, which ensures that design documents are consistent and that design requirements for IROFS are met. During construction and testing, this verification also extends to verification that as-built configurations are consistent with the design, and that testing that is specified to demonstrate performance of IROFS is accomplished successfully. Periodic audits and assessments of the configuration management program and of the design confirm that the system meets its goals and that the design is consistent with the design bases. The corrective action process occurs in accordance with the LES QA Program and associated procedures in the event problems are identified. Prompt corrective actions are developed as a result of incident investigations or in response to audit or assessment results.

11.1.1.4 Description of Configuration Management Activities

Configuration management includes those activities conducted under design control provisions for ensuring that design and construction documentation is prepared, reviewed, and approved in accordance with a systematic process. This process includes interdisciplinary reviews appropriate to ensure consistency between the design and the design bases of IROFS. During construction, it also includes those activities that ensure that construction is consistent with design documents. Finally, it includes activities that provide for operation of the IROFS in accordance with the limits and constraints established in the ISA, and that provide for control of changes to the facility in accordance with 10 CFR 70.72 (CFR, 2003e).

Configuration management also includes records to demonstrate that personnel conducting activities that are relied on for safety or that are associated with IROFS are appropriately qualified and trained to conduct that work.

Implementing documents are controlled within the document control system. These documents support configuration management by ensuring that only reviewed and approved procedures, specifications and drawings are used for procurement, construction, installation, testing, operation, and maintenance of IROFS, as appropriate.

11.1.1.5 Organizational Structure and Staffing Interfaces

The configuration management program is administered by the Engineering and Contracts organization during design and construction. Engineering includes engineering disciplines with responsible lead engineers in charge of each discipline, under the direction of design managers or project managers who report to the Engineering and Contracts Manager. The lead discipline engineers have primary technical responsibility for the work performed by their disciplines, and are responsible for the conduct of interdisciplinary reviews as discussed previously in this section. Reviews are also conducted, as appropriate, by construction management, operations, QA, and procurement personnel. The design control process also interfaces with the document control and records management process via procedures.

The various LES departments and contractors of LES perform quality-related activities. The primary LES contractors are responsible for development of their respective QA Programs, which shall be consistent with the requirements of the LES QA Program for those activities determined to be within the scope of the LES QA Program. The interfaces between contractors and LES or among contractors shall be documented. LES and contracted personnel have the responsibility to identify quality problems. If a member of another area disagrees, that individual is instructed to take the matter to appropriate management. The disagreement may either be resolved at this level or at any level up to and including the LES President.

11.1.2 Design Requirements

Design requirements and associated design bases are established and maintained by the Engineering and Contracts organization during design and construction and by the Technical Services organization during operations. The configuration management controls on design requirements and the integrated safety analysis of the design bases are described previously in this section. Design requirements are documented in a design requirements document that provides for a hierarchical distribution of these requirements through basis of design documents. The design requirements document and basis of design documents are controlled under the design control provisions of the configuration management program as described above, and are subject to the same change control as analyses, specifications, and drawings. Computer codes used in the design of IROFS are also subject to these design control measures, with additional requirements as appropriate for software control, verification, and validation.

IROFS, any items that affect the function of the IROFS, and, in general, items required to satisfy regulatory requirements are designated as QA Level 1. The associated design documents are subject to interdisciplinary reviews and design verification. Analyses constituting the integrated safety analysis of the design bases are subject to the same requirements. Changes to the design are evaluated to ensure consistency with the design bases.

IROFS are listed in the design requirements document. This list will be augmented and maintained current as appropriate as IROFS are identified in more detail during detailed design.

A qualified individual who specifies and includes the appropriate codes, standards, and licensing commitments within the design documents prepares each design document, such as a calculation, specification, procedure, or drawing. This individual also notes any deviations or changes from such standards within the design documentation package. Each design document is then checked by another individual qualified in the same discipline and is reviewed for concept and conformity with the design inputs. These design inputs are in sufficient detail to permit verification of the document. The manager having overall responsibility for the design function approves the document. The Engineering Manager documents the entire review process in accordance with approved procedures. These procedures include provisions to assure that appropriate quality standards are specified in design documents, including quantitative or qualitative acceptance criteria. The QA Director conducts audits on the design control process using independent technically qualified individuals to augment the QA audit team.

During the check and review, emphasis is placed on assuring conformance with applicable codes, standards and license application design commitments. The individuals in engineering assigned to perform the check and review of a document have full and independent authority to withhold approval until questions concerning the work have been resolved. Design reviews, alternative calculations, or qualification testing accomplishes verification of design. The bases for a design, such as analytical models, theories, examples, tables, codes and computer programs must be referenced in the design document and their application verified during check and review. Model tests, when required to prove the adequacy of a concept or a design, are reviewed and approved by the responsible qualified individual. Testing used for design verification shall demonstrate adequacy of performance under conditions that simulate the most adverse design conditions. The tests used for design verification must meet all the design requirements.

Qualified individuals other than those who performed the design but may be from the same organization perform design verification. Verification may be performed by the supervisor of the individual performing the design, provided this need is documented, approved in advance by the supervisor's management, and the supervisor did not specify a singular design approach or rule out certain design considerations, and did not establish the design inputs used in the design or, provided the supervisor is the only individual in the organization competent to perform the verification. The verification by a supervisor of their own design constraints, design input, or design work would only occur in rare instances. This would occur only when the supervisor is the only individual in the organization competent to perform the verification. These instances are authorized and documented in writing on a case-by-case basis.

Independent design verification shall be accomplished before the design document (or information contained therein) is used by other organizations for design work or to support other activities such as procurement, construction, or installation. When this is not practical due to time constraints, the unverified portion of the document is identified and controlled. In all cases, the design verification shall be completed before relying on the item to perform its function or installation becomes irreversible. Any changes to the design and procurement documents, including field changes, must be reviewed, checked and approved commensurate with the original approval requirements.

After design documents have been properly prepared, checked, reviewed, and approved by the appropriate parties, the responsible engineer sends the document to document control for distribution. When required, each recipient of a design document verifies receipt of such document to the document control center.

The document control center, after verification of distribution to a recipient, maintains the required documentation in its files.

When deficiencies are identified which affect the design of IROFS, such deficiencies are documented and resolved in accordance with approved CAP procedures. In accordance with the CAP the report is forwarded for appropriate review to the responsible manager, who coordinates further review of the problem and revises all design documents affected by the deficiency as necessary. Where required, the responsible manager forwards the report to the engineers in other areas, who coordinate necessary revisions to their affected documents

Design interfaces are maintained by communication among the principals. Methods by which this is accomplished include the following:

A. Design documents are reviewed by the responsible engineer or authorized representative. As appropriate, subsequent review or waiver of review by the other area engineers is documented.

B. Project review meetings are scheduled and held to coordinate design, procurement, construction and pre-operational testing of the facility. These meetings provide a primary working interface among the principal organizations.

C. Reports of nonconformances are transmitted and controlled by procedures. As required by the nonconformance procedure, the QA Director/Manager or designee approves resolution of nonconformances.

During the operational phase, measures are provided to ensure responsible facility personnel are made aware of design changes and modifications that may affect the performance of their duties.

11.1.2.1 Configuration Management Controls on the Design Requirements

Configuration control is accomplished during design through the use of procedures for controlling design, including preparation, review (including interdisciplinary review and preparation of NCS analyses and NCS evaluations as applicable), and design verification where appropriate, approval, and release and distribution for use. Engineering documents are assessed for QA level classification. Changes to the approved design also are subject to a review to ensure consistency with the design bases of IROFS.

Configuration verification is also accomplished through design verification, which ensures that design documents are consistent and that design requirements for IROFS are met. During construction and testing, this verification also extends to verification that as-built configurations are consistent with the design, and that testing that is specified to demonstrate performance of IROFS is accomplished successfully.

The QA Program requires procedures that specify that work performed shall be accomplished in accordance with the requirements and guidelines imposed by applicable specifications, drawings, codes, standards, regulations, quality assurance criteria and site characteristics.

Acceptance criteria established by the designer are incorporated in the instructions, procedures and drawings used to perform the work. Documentation is maintained, including test results, and inspection records, demonstrating that the work has been properly performed. Procedures also provide for review, audit, approval and documentation of activities affecting the quality of items to ensure that applicable criteria have been met.

Maintenance, modification, and inspection procedures are reviewed by qualified personnel knowledgeable in the quality assurance disciplines to determine:

A. The need for inspection, identification of inspection personnel, and documentation of inspection result

B. That the necessary inspection requirements, methods, and acceptance criteria have been identified.

Facility procedures shall be reviewed by an individual knowledgeable in the area affected by the procedure on a frequency determined by the age and use of the procedure to determine if changes are necessary or desirable. Procedures are also reviewed to ensure procedures are maintained up-to-date with facility configuration. These reviews are intended to ensure that any modifications to facility systems, structures or components are reflected in current maintenance, operations and other facility procedures.

11.1.3 Document Control

Procedures are established which control the preparation and issuance of documents such as manuals, instructions, drawings, procedures, specifications, procurement documents and supplier-supplied documents, including any changes thereto. Measures are established to ensure documents, including revisions, are adequately reviewed, approved, and released for use by authorized personnel.

Document control procedures require documents to be transmitted and received in a timely manner at appropriate locations including the location where the prescribed activity is to be performed. Controlled copies of these documents and their revisions are distributed to and used by the persons performing the activity.

Superseded documents are destroyed or are retained only when they have been properly labeled. Indexes of current documents are maintained and controlled.

Document control is implemented in accordance with procedures. An electronic document management system is used both to file project records and to make available the latest revision (i.e., the controlled copy) of design documents. The system provides an "official" copy of the current document, and personnel are trained to use this system to retrieve controlled documents. The system is capable of generating indices of controlled documents, which are uniquely numbered (including revision number). Controlled documents are maintained until cancelled or superseded, and cancelled or superseded documents are maintained as a record, currently for the life of the project or termination of the license, whichever occurs later. Hard-copy distribution of controlled documents is provided when needed in accordance with applicable procedures (e.g., when the electronic document management system is not available).

A part of the configuration management program, the document control and records management procedures, as appropriate, capture the following documents:

- Design requirements, through the controlled copy of the design requirements document
- The design bases, through the controlled copy of the basis of design documents
- The integrated safety analysis of the design bases of IROFS, through the controlled copies of supporting analyses
- Nuclear Criticality Safety Analyses
- Nuclear Criticality Safety Evaluations
- As-built drawings
- Specifications
- All procedures that are IROFS
- Procedures involving training

- QA
- Maintenance
- Audit and assessment reports
- Emergency operating procedures
- Emergency response plans
- System modification documents
- Assessment reports
- Engineering documents including analyses, specifications, technical reports, and drawings.

These items are documented in approved procedures.

11.1.4 Change Control

Procedures control changes to the technical baseline. The process includes an appropriate level of technical, management, and safety review and approval prior to implementation. During the design phase of the project, the method of controlling changes is the design control process described in the QA Program. This process includes the conduct of interdisciplinary reviews that constitute a primary mechanism for ensuring consistency of the design with the design bases. During both construction and operation, appropriate reviews to ensure consistency with the design bases of IROFS and the ISA, respectively, will similarly ensure that the design is constructed and operated/modified within the limits of the design basis. Additional details are provided below.

11.1.4.1 Design Phase

Changes to the design include a systematic review of the design bases for consistency. In the event of changes to reflect design or operational changes from the established design bases, both the integrated safety analysis and other documents affected by design bases of IROFS including the design requirements document and basis of design documents, as applicable are properly modified, reviewed, and approved prior to implementation. Approved changes are made available to personnel through the document control function discussed previously in this section.

During design (i.e., prior to issuance of the NEF Materials License), the method of ensuring consistency between documents, including consistency between design changes and the safety assessment, is the interdisciplinary review process. The interdisciplinary reviews ensure design changes either (1) do not impact the ISA, (2) are accounted for in subsequent changes to the ISA, or (3) are not approved or implemented. Prior to issuance of the License, LES will notify the NRC of potential changes that reduce the level of commitments or margin of safety in the design bases of IROFS.

11.1.4.2 Construction Phase

When the project enters the construction phase, changes to documents issued for construction, fabrication, and procurement will be documented, reviewed, approved, and posted against each affected design document. Vendor drawings and data also undergo an interdisciplinary review to ensure compliance with procurement specifications and drawings, and to incorporate interface requirements into facility documents.

During construction, design changes will continue to be evaluated against the approved design bases. Changes are expected to the design as detailed design progresses and construction begins. A systematic process consistent with the process described above will be used to evaluate changes in the design against the design bases of IROFS and the ISA. Upon issuance of the NEF Materials License, the configuration change process will fully implement the provisions of 10 CFR 70.72 (CFR, 2003e), including reporting of changes made without prior NRC approval as required by 10 CFR 70.72(d)(2) and (3). Any change that requires Commission approval, will be submitted as a license amendment request as required by 10 CFR 70.72(d)(1) and the change will not be implemented without prior NRC approval.

11.1.4.3 Operations Phase

During the operations phase, changes to design will also be documented, reviewed, and approved prior to implementation. LES will implement a change process that fully implements the provisions of 10 CFR 70.72 (CFR, 2003e). Measures are provided to ensure responsible facility personnel are made aware of design changes and modifications that may affect the performance of their duties.

In order to provide for the continued safe and reliable operation of the facility structures, systems and components, measures are implemented to ensure that the quality of these structures, systems and components is not compromised by planned changes (modifications). After issuance of the Operating License, the Plant Manager is responsible for the design of and modifications to facility structures, systems or components. The design and implementation of modifications are performed in a manner so as to assure quality is maintained in the remainder of the system that is being modified, or as dictated by applicable regulations.

The administrative instructions for modifications are contained in a facility administrative procedure that is approved, including revisions, by the Technical Services Manager with concurrence of the Quality Assurance Manager. The modification procedure contains the following items necessary to ensure quality in the modification program:

- The requirements that shall be met to implement a modification
- The requirements for initiating, approving, monitoring, designing, verifying, and documenting modifications. The facility modification procedure shall be written to ensure that policies are formulated and maintained to satisfy the quality assurance requirements specified in the LES QA Program, as applicable.

Each change to the facility or to activities of personnel shall have an evaluation performed in accordance with the requirements of 10 CFR 70.72 (CFR, 2003e), as applicable. Each modification shall also be evaluated for any required changes or additions to the facility's procedures, personnel training, testing program, or regulatory documents.

For any change (i.e., new design or operation, or modification to the facility or to activities of personnel, e.g., site structures, systems, components, computer programs, processes, operating procedures, management measures) that involves or could affect uranium on site, an NCS evaluation and, if required, an NCS analysis shall be prepared and approved. Prior to implementing the change, it shall be determined that the entire process will be subcritical (with applicable margin for safety) under both normal and credible abnormal conditions.

Each modification is also evaluated and documented for radiation exposure to minimize worker exposures in keeping with the facility ALARA program, criticality and worker safety requirements and/or restrictions. Other areas of consideration in evaluating modifications may include, but are not limited to the review of:

- Modification cost
- Lessons learned from similar completed modifications
- QA aspects
- Potential operability or maintainability concerns
- Constructability concerns
- Post-modification testing requirements
- Environmental considerations
- Human factors.

After completion of a modification to a structure, system, or component, the modification Project Manager, or designee, shall ensure that all applicable testing has been completed to ensure correct operation of the system(s) affected by the modification and documentation regarding the modification is complete. In order to ensure operators are able to operate a modified system safely, when a modification is complete, all documents necessary, e.g., the revised process description, checklists for operation and flowsheets are made available to operations and maintenance departments once the modified system becomes "operational." Appropriate training on the modification is completed before a system is placed in operation. A formal notice of a modification being completed is distributed to all appropriate managers. As-built drawings incorporating the modification are completed promptly. These records shall be identifiable and shall be retained for the duration of the facility license.

11.1.5 Assessments

Periodic assessments of the configuration management program are conducted to determine the system's effectiveness and to correct deficiencies. These assessments include review of the adequacy of documentation and system walk downs of the as-built facility. Such audits and assessments are conducted and documented in accordance with procedures and scheduled as discussed in Appendix A, Section 18, "Audit Schedules."

Periodic audits and assessments of the configuration management program and of the design confirm that the system meets its goals and that the design is consistent with the design bases. Incident investigations occur in accordance with the QA Program and associated CAP procedures in the event problems are encountered. Prompt corrective actions are developed as a result of incident investigations or in response to adverse audit/assessment results, in accordance with CAP procedures.

11.2 MAINTENANCE

This section outlines the maintenance and functional testing programs to be implemented for the operations phase of the facility. Preventive maintenance activities, surveillance, and performance trending provide reasonable and continuing assurance that IROFS will be available and reliable to perform their safety functions.

The purpose of planned and scheduled maintenance for IROFS is to ensure that the equipment and controls are kept in a condition of readiness to perform the planned and designed functions when required. Appropriate plant management is responsible for ensuring the operational readiness of IROFS under this control. For this reason, the maintenance function is administratively closely coupled to operations. The Maintenance organization plans, schedules, tracks, and maintains records for maintenance activities.

In order to provide for the continued safe and reliable operation of the facility structures, systems and components, measures are implemented to ensure that the quality of these structures, systems and components is not compromised by planned changes (modifications) or maintenance activities. After issuance of the Operating License, the Plant Manager is responsible for the design of and modifications to facility structures, systems or components and all maintenance activities. The design and implementation of modifications are performed in a manner so as to assure quality is maintained in a manner commensurate with the remainder of the system which is being modified, or as dictated by applicable regulations.

The administrative instructions for modifications are contained in a facility administrative procedure that is approved, including revisions, by the Technical Services Manager with concurrence of the Quality Assurance Manager. The modification procedure contains the following items necessary to ensure quality in the modification program:

- The requirements which shall be met to implement a modification
- The requirements for initiating, approving, monitoring, designing, verifying, and documenting modifications. The facility modification procedure shall be written to ensure that policies are formulated and maintained to satisfy the quality assurance standards specified in the LES QA Program, as applicable.

Listed below are methods or practices that will be applied to the corrective, preventive, and functional-test maintenance elements. LES will prepare written procedures for performance of these methods and practices. These methods and practices include, as applicable:

Authorized work instructions with detailed steps and a reminder of the importance of the IROFS identified in the ISA Summary:

- Parts lists
- As-built or redlined drawings
- A notification step to the Operations function before conducting repairs and removing an IROFS from service

- Radiation Work Permits
- Replacement with like-kind parts and the control of new or replacement parts to ensure compliance with 10 CFR 21 (CFR, 2003a)
- *Compensatory measures while performing work on IROFS*
- Procedural control of removal of components from service for maintenance and for return to service
- Ensuring safe operations during the removal of IROFS from service
- Notification to Operations personnel that repairs have been completed.

Written procedures for the performance of maintenance activities include the steps listed above. The details of maintenance procedure acceptance criteria, reviews, and approval are provided in Section 11.4, Procedures Development and Implementation.

As applicable, contractors that work on or near IROFS identified in the ISA Summary will be required by LES to follow the same maintenance procedures described for the corrective, preventive, functional testing, or surveillance/monitoring activities listed above for the maintenance function.

Maintenance procedures involving IROFS commit to the topics listed below for corrective and preventive maintenance, functional testing after maintenance, and surveillance/monitoring maintenance activities:

- Pre-maintenance activities require reviews of the work to be performed, including procedure reviews for accuracy and completeness.
- New procedures or work activities that involve or could affect uranium on site require preparation and approval of an NCS evaluation and, if required, an NCS analysis.
- Steps that require notification of all affected parties (operators and appropriate managers) before performing work and on completion of maintenance work. The discussion includes potential degradation of IROFS during the planned maintenance.
- Control of work by comprehensive procedures to be followed by maintenance technicians. Maintenance procedures are reviewed by the various safety disciplines, including criticality, fire, radiation, industrial, and chemical process safety. The procedures describe, as a minimum, the following:
  - Qualifications of personnel authorized to perform the maintenance, functional testing or surveillance/monitoring
  - Controls on and specification of any replacement components or materials to be used (this will be controlled by Configuration Management, to ensure like-kind replacement and adherence to 10 CFR 21 (CFR, 2003a))
  - Post-maintenance testing to verify operability of the equipment
  - Tracking and records management of maintenance activities

- Safe work practices (e. g., lockout/tag out, confined space entry, moderation control or exclusion area, radiation or hot work permits, and criticality, fire, chemical, and environmental issues).

Maintenance activities generally fall into the following categories:

- Surveillance/monitoring
- Corrective maintenance
- Preventive maintenance
- Functional testing.

These maintenance categories are discussed in the following sections.

11.2.1 Surveillance/Monitoring

Surveillance/monitoring is utilized to detect degradation and adverse trends of IROFS so that action may be taken prior to component failure. The monitored parameters are selected based upon their ability to detect the predominate failure modes of the critical components. Data sources include; surveillance, periodic and diagnostic test results, plant computer information, operator rounds, walk downs, as-found conditions, failure trending, and predictive maintenance. Surveillance/monitoring and reporting is required for SSC that are identified as IROFS and any SSC and administrative controls that could impact the functions of an IROFS.

Plant performance criteria are established to monitor plant performance and to monitor IROFS functions and component parameters. These criteria are established using Urenco industry experience, operating data, surveillance data, and plant equipment operating experience. These criteria ensure the reliability and availability of IROFS. The performance criteria are also used to demonstrate that the performance or condition of an IROFS is being effectively controlled through appropriate predictive and repetitive maintenance strategies so that IROFS remain capable of performing their intended function.

Surveillance of IROFS is performed at specified intervals. The purpose of the surveillance program is to measure the degree to which IROFS meet performance specifications. The results of surveillances are trended, and when the trend indicates potential IROFS performance degradation, preventive maintenance frequencies are adjusted or other appropriate corrective action is taken.

Incident investigations may identify root causes of failures that are related to the type or frequency of maintenance. The lessons learned from such investigations are factored into the surveillance/monitoring and preventive maintenance programs as appropriate.

Maintenance procedures prescribe compensatory measures, if appropriate, for surveillance tests of IROFS that can be performed only while equipment is out of service.

Records showing the current surveillance schedule, performance criteria, and test results for all IROFS will be maintained in accordance with the Record Management System.

Results of surveillance/monitoring activities related to IROFS via the configuration management program will be evaluated by all safety disciplines to determine any impact on the ISA and any updates needed.

11.2.2 Corrective Maintenance

Corrective maintenance involves repair or replacement of equipment that has unexpectedly degraded or failed. Corrective maintenance of IROFS restores the equipment to acceptable performance through a planned, systematic, controlled, and documented approach for the repair and replacement activities.

Following any corrective maintenance on IROFS, and before returning an IROFS to operational status, functional testing of the IROFS, if necessary, is performed to ensure the IROFS performs its intended safety function.

The CAP requires facility personnel to determine the cause of conditions adverse to quality and promptly act to correct these conditions.

Results of corrective maintenance activities related to IROFS via the configuration management program will be evaluated by all safety disciplines to determine any impact on the ISA and any updates needed.

11.2.3 Preventive Maintenance

Preventive maintenance (PM) includes preplanned and scheduled periodic refurbishment, partial or complete overhaul, or replacement of IROFS, if necessary, to ensure their continued safety function. Planning for preventive maintenance includes consideration of results of surveillance and monitoring, including failure history. PM also includes instrument calibration and testing.

The PM program procedures and calibration standards (traceable to the national standards system or to nationally accepted calibration techniques, as appropriate) enable the facility personnel to calibrate equipment and monitoring devices important to plant safety and safeguards. Testing performed on IROFS that are not redundant will provide for compensatory measures to be put into place to ensure that the IROFS function is performed until it is put back into service.

Urenco's extensive experience in the industry (30 years) is used to determine initial PM frequencies and procedures. In determining the frequency of PM, consideration is given to appropriately balancing the objective of preventing failures through maintenance against the objective of minimizing unavailability of IROFS because of PM. In addition, feedback from PM and corrective maintenance and the results of incident investigations and identified root causes are used, as appropriate, to modify the frequency or scope of PM. The rationale for deviations from industry standards or vendor recommendations for PM shall be documented.

After conducting preventive maintenance on IROFS, and before returning an IROFS to operational status, functional testing of the SSC, if necessary, is performed to ensure the IROFS performs its intended safety function. Functional testing is described in detail in Section 11.2.4, Functional Testing.

All records pertaining to preventive maintenance will be maintained in accordance with the Records Management System.

Results of preventive maintenance activities related to IROFS via the configuration management system will be evaluated by all safety disciplines to determine any impact on the ISA and any updates needed.

11.2.4 Functional Testing

Functional testing of IROFS is performed as appropriate following initial installation, as part of periodic surveillance testing, and after corrective or preventive maintenance or calibration to ensure that the item is capable of performing its safety function when required.

The overall testing program is broken into the two major testing programs and within each testing program are two testing categories:

A. Preoperational Testing Program
    1. Functional Testing
    2. Initial Startup Testing.

B. Operational Testing Program
    1. Periodic Testing
    2. Special Testing.

Results of surveillance/monitoring activities related to IROFS via the configuration management program will be evaluated by all safety disciplines to determine any impact on the ISA and any updates needed.

11.2.4.1 Objectives

The objectives of the overall facility preoperational and operational testing programs are to ensure that items relied on for safety:

A. Have been adequately designed and constructed

B. Meet contractual, regulatory, and licensing requirements

C. Do not adversely affect worker or the public health and safety

D. Can be operated in a dependable manner so as to perform their intended function.

Additionally, the preoperational and operational testing programs ensure that operating and emergency procedures are correct and that personnel have acquired the correct level of technical expertise.

Periodic testing at the facility consists of that testing conducted on a periodic basis to monitor various facility parameters and to verify the continuing integrity and capability of IROFS.

Special testing at the facility consists of that testing which does not fall under any other testing program. This testing is of a non-recurring nature and is intended to enhance or supplement existing operational testing rather than replace or supersede other testing or testing programs.

11.2.4.2 Procedure Content

Test Procedures are sufficiently detailed that qualified personnel can perform the required functions without direct supervision. The content of test procedures is uniform to the extent practicable and consists of the following:

A. Title

Each procedure contains a title descriptive of the activities to which it applies.

B. Purpose

The purpose for which the procedure is intended is stated. This statement of applicability is as clear and concise as practicable.

C. References

References are made to specific material used in the preparation and performance of a procedure. This includes applicable drawings, instruction manuals, specifications, and sections of the facility's operating license. These references are listed in a manner as to allow ready location of the material.

D. Time Required

As applicable, estimates of the manpower and time requirements for performance of the specified testing activity are indicated.

E. Prerequisites

Each procedure specifies those items that are required to be completed prior to the performance of the specified testing (e.g., a previous test or special operating conditions). This listing also includes any tests that are to be performed concurrently with the specified testing. Provisions are made to document verification of the completion of the specified prerequisite tests.

F. Test Equipment

Each procedure contains a listing of special test equipment required in performing the specified testing. Procedures contain information and/or references for the items listed such as instruction manuals or procedures.

G. Limits and Precautions

Limits on parameters being controlled and corrective measures necessary to return a parameter to its normal control band are specified. Procedures specifically incorporate limits and corrective measures for all operations affecting criticality safety.

Precautions are specified which alert the individual performing the task, of those situations for which important measures need to be taken early, or where extreme care must be used to protect personnel and equipment or to avoid an abnormal or an emergency situation.

H. Required Plant Unit Status

The procedure specifies the plant unit status necessary to perform the specified testing. Provisions are made to document compliance with the status specified.

I. Prerequisite System Conditions

The procedure specifies the prerequisite system conditions necessary to perform the specified testing. Provisions are made to document compliance with the conditions specified.

J. Test Method

Each procedure contains a brief descriptive section that summarizes the method to be used for performing the specified testing.

K. Data Required

Each procedure specifies any data that must be compiled in the performance of the specified testing in order to verify satisfactory completion of the specified testing. This includes a description of any calculations necessary to reduce raw data to a workable form.

L. Acceptance Criteria

Each procedure states the criteria for evaluating the acceptability of the results of the specified testing. Test results are reduced to a meaningful and readily understandable form in order to facilitate evaluation of their acceptability. Adequate provisions are made to allow documentation of the acceptability, or unacceptability, of test results.

M. Procedure

Procedures contain step-by-step directions in the degree of detail necessary for performing the required testing. References to documents other than the subject procedure are included, as applicable. However, references are identified within these step-by-step directions when the sequence of steps requires that other tasks (not specified by the subject procedure) be performed prior to or concurrent with a procedure step. Where witnessing of a test is required, adequate provisions are made in the test procedure to allow for the required witnessing and to document the witnessing. Cautionary notes, applicable to specific steps, are included and are distinctly identified.

N. Enclosures

Data sheets, checklists and diagrams are attached to the procedure. In particular, checklists utilized to avoid or simplify lengthy or complex procedures are attached as enclosures.

11.2.4.3 Preoperational Testing Program

Preoperation functional tests are completed prior to UF_6 introduction. Other preoperational tests, not required prior to UF_6 introduction and not related to IROFS, such as office building ventilation tests, may be completed following UF_6 introduction. Tests (or portions of tests), which are not required to be completed before UF_6 introduction are identified in the test plan.

The Preoperational testing program comprises three parts:

- Constructor turnover
- Preoperational functional testing
- Initial start up testing.

Constructor Turnover

The constructor is responsible for completion of all as-built drawing verification, purging, cleaning, vacuum testing, system turnover and initial calibration of instrumentation in accordance with design and installation specifications provided by the architect engineers and vendors. As systems or portions of systems are turned over to LES, preoperational testing shall begin. The Technical Services Manager is responsible for coordination of the preoperational and startup test program.

The preoperational test plan including test summaries for all systems is available to the NRC at least 90 days prior to the start of testing. Subsequent changes to the preoperational test plan are also made available to the NRC. Preoperational testing as a minimum includes all system or component tests required by the pertinent design code which were not performed by the constructor prior to turnover. In addition, preoperational tests include all testing necessary to demonstrate that the IROFS are capable of performing their intended function.

Functional Testing

Preoperational functional testing at the facility consists of that testing conducted to initially determine various facility parameters and to initially verify the capability of SSC to meet performance requirements. The tests conducted are primarily associated with IROFS (QA Level 1) and certain QA Level 2 structures, systems and components, but may also include a number of other tests of a technical or financial interest to LES.

Preoperational functional tests are performed following constructor turnover. The major objective of preoperational functional testing is to verify that IROFS essential to the safe operation of the plant are capable of performing their intended function.

For structures, systems and components that are not QA Level 1, acceptance criteria are established to ensure worker-safety Occupational Safety and Health Administration (OSHA), reliable and efficient operation of the system and to demonstrate the performance of intended functions.

Initial startup testing at the facility consists of that testing which includes initial UF_6 introduction and all subsequent testing through the completion of Enrichment Setting Verification for each cascade. "Enrichment Setting Verification" is the verification of a selected enrichment weight percent by measurement of a physical sample collected during the "Enrichment Setting Verification" test run.

Initial startup testing is performed beginning with the introduction of UF_6 and ending with the start of commercial operation. The purpose of initial startup testing is to ensure safe and orderly UF_6 feeding and to verify parameters assumed in the ISA. Examples of initial startup tests include passivation and the filling phase.

Records of the preoperational and startup tests required prior to operation are maintained. These records include testing schedules and the testing results for all IROFS.

Initial Startup Testing

All aspects of initial startup testing are conducted under appropriate test procedures. See Section 11.4, Procedures Development and Implementation, for a detailed description of facility procedures. The use of properly reviewed and approved test procedures is required for all preoperational and startup tests. The results of each preoperational test are reviewed and approved by the responsible department manager or designee before they are used as the basis of continuing the test program. The results of startup testing are reviewed and approved by the Technical Services Manager. In addition, the results of each individual startup test will receive the same review as that described for preoperation functional tests. All modifications to IROFS that are found necessary are subjected to an evaluation per 10 CFR 70.72 (CFR, 2003e) prior to making the change.

The impact of modifications on future and completed testing is evaluated during the 10 CFR 70.72 (CFR, 2003e) evaluation process and retesting is conducted as required.

Copies of approved test procedures are made available to NRC personnel approximately 60 days prior to their intended use, and not less than 60 days prior to the scheduled introduction of UF_6 for startup tests.

The overall preoperational functional testing program is reviewed, prior to initial UF_6 introduction, by the Plant Manager and all department Managers to ensure that all prerequisite testing is complete.

The facility operating, emergency and surveillance procedures are use-tested throughout the testing program phases and are also used in the development of preoperation functional testing and initial startup testing procedures to the extent practicable. The trial use of operating procedures serves to familiarize operating personnel with systems and plant operation during the testing phases and also serves to ensure the adequacy of the procedures under actual or simulated operating conditions before plant operation begins.

Procedures which cannot be use-tested during the testing program phase are revised based on initial use-testing, operating experience and comparison with the as-built systems. This ensures that these procedures are as accurate and comprehensive as practicable.

11.2.4.4 Operational Testing Program

The operational testing program consists of periodic testing and special testing. Periodic testing is conducted at the facility to monitor various facility parameters and to verify the continuing integrity and capability of facility IROFS. Special testing which may be conducted at the facility is testing which does not fall under any other testing program and is of a non-recurring nature.

The Maintenance Manager has overall responsibility for the development and conduct of the operational testing program and in conjunction with the Health, Safety and Environment (HS&E) Manager ensures that all testing commitments and applicable regulatory requirements are met.

The HS&E Manager shall ensure that new surveillance requirements or testing commitments are identified to the Maintenance Manager. The Maintenance Manager shall make responsibility assignments for new testing requirements.

Surveillance commitments, procedures identified to satisfy these commitments and surveillance procedure responsibility assignments for the facility are identified in a computer database. The database is also used to ensure surveillance testing is completed in the required time interval for all departments.

Test Coordinators are also used for operational testing. The Test Coordinator has the responsibility to be thoroughly familiar with the procedure to be performed. The Test Coordinator should have an adequate period of time in which to review the procedure and the associated system before the start of the test. It is the responsibility of the appropriate section or department head to designate and ensure that each Test Coordinator meets the appropriate requirements. Operational testing is usually performed by each shift. The Test Coordinator, as part of the shift personnel, also performs regular shift duties in performance of the tests.

The Test Coordinator has the following responsibilities regarding the conduct of testing:

A. Verification of all system and plant unit prerequisites

B. Observance of all limits and precautions during the conduct of the test

C. Compliance with the requirements of the facility license and any other facility directives regarding procedure changes and documentation

D. Identifying and taking corrective actions necessary to resolve system deficiencies or discrepancies observed during the conduct of the test

E. Verification of proper data acquisition, evaluation or results, and compliance with stated acceptance criteria

F. Ensuring that adequate personnel safety precautions are observed during the conduct of the test

G. Coordinating and observing additional manpower and support required from other departments or organizations.

Periodic and special testing procedures are sufficiently detailed that qualified personnel can perform the required functions without direct supervision. The administration requirements for periodic and special testing procedures are the same as ones used for preoperational functional test and initial startup test procedures as identified in Section 11.2.4.3, Preoperational Testing Program. Spaces for initials and dates are required for the following sections:

A. Prerequisite Tests

B. Required Facility (or Plant Unit) Status

C. Prerequisite System Conditions

D. Procedure

E. Enclosures (where calculations are made).

Whenever possible generic procedures and enclosures for recording data for periodic and special tests are used. Also whenever possible, the enclosure is designed as a self-sufficient document that can be filed as evidence that the subject test was performed. Enclosures used as self-sufficient documents should contain sign-off blanks (Initials/Date) to verify that prerequisite tests, required facility status and prerequisite facility or plant unit status and prerequisite system conditions are met before conduct of the test.

11.2.4.4.1 Periodic Testing

The periodic testing program at the facility consists of testing conducted on a periodic basis to verify the continuing capability of IROFS to meet performance requirements.

The facility periodic test program verifies that the facility:

A. Complies with all regulatory and licensing requirements

B. Does not endanger health and minimizes danger to life or property

C. Is capable of operation in a dependable manner so as to perform its intended function.

The facility periodic testing program begins during the preoperational testing stage and continues throughout the facility's life.

A periodic testing schedule is established to ensure that all required testing is performed and properly evaluated on a timely basis. The schedule is revised periodically, as necessary, to reflect changes in the periodic testing requirements and experience gained during plant operation. Testing is scheduled such that the safety of the plant is never dependent on the performance of an IROFS that has not been tested within its specified testing interval.

Periodic test scheduling is handled through the Maintenance department. The Maintenance department maintains the periodic test status index on the computer database. The purpose of this index is to assist groups in assuring that all surveillances are being completed within the required test interval.

The database includes all periodic testing, calibration or inspection required by regulatory requirements or licensing commitments, and provides the following information for each surveillance:

- Test #
- Title
- Equipment #
- Work Request # (if applicable)
- Test Frequency
- Plant Cascade #
- Last date test was performed
- Next date test is due.

In the event that a test cannot be performed within its required interval due to system or plant unit conditions, the responsible department notifies in writing, on the applicable form, the HS&E Manager, Operations Manager, and the Maintenance Manager, as appropriate. The originating department retains a copy as a record of the transmittal. The responsible department lists the earliest possible date the test could be performed and the latest date along with the required system or unit-mode condition. However, the responsible department will ensure that the test is performed as soon as practical once required conditions are met, regardless of the estimated date given earlier.

Periodic testing and surveillance associated with QA Level 1 and 2 structures, systems and components are performed in accordance with written procedures.

11.2.4.4.2 Special Testing

Special testing is testing conducted at the facility that is not a facility preoperational test, periodic test, post-modification test, or post-maintenance test. Special testing is of a non-recurring nature and is conducted to determine facility parameters and/or to verify the capability of IROFS to meet performance requirements. Purposes of special testing include, but are not necessarily limited to, the following:

A. Acquisition of particular data for special analysis

B. Determination of information relating to facility incidents

C. Verification that required corrective actions reasonably produce expected results and do not adversely affect the safety of operations

D. Confirmation that facility modifications reasonably produce expected results and do not adversely affect systems, equipment and/or personnel by causing them to function outside established design conditions; applicable to testing performed outside of a post-modification test.

The determination that a certain plant activity is a Special Test is intended to exclude those plant activities which are routine surveillances, normal operational evolutions, and activities for which there is previous experience in the conduct and performance of the activity. At the discretion of the Plant Manager, any test may be conducted as a special test. In making this determination, facility management includes the following evaluations of characteristics of the activity:

A. Does the activity involve an unusual operational configuration for which there is no previous experience?

B. Does the activity have the propensity, if improperly conducted, to significantly affect primary plant parameters?

C. Does the activity involve seldom-performed evolutions, meeting one of the above criteria, in which the time elapsed since the previous conduct of the activity renders prior experience not useful?

11.3 TRAINING AND QUALIFICATIONS

This section describes the training program for the operations phase of the facility, including preoperational functional testing and initial startup testing. The training program requirements apply to those plant personnel who perform activities relied on for safety.

The QA Program provides training and qualification requirements, during the design, construction, and operations phases, for QA training of personnel performing QA levels 1 and 2 work activities; for nondestructive examination, inspection, and test personnel; and for QA auditors.

The principle objective of the LES training program system is to ensure job proficiency of all facility personnel involved in work through effective training and qualification. The training program system is designed to accommodate future growth and meet commitments to comply with applicable established regulations and standards.

Qualification is indicated by successful completion of prescribed training, demonstration of the ability to perform assigned tasks and where required by regulation, maintaining a current and valid license issued by the agency establishing the requirements. Training is designed, developed and implemented according to a systematic approach. Employees are provided with formal training to establish the knowledge foundation and on-the-job training to develop work performance skills. Continuing training is provided, as required, to maintain proficiency in these knowledge and skill components, and to provide further employee development.

The training program described in this section is consistent with that previously submitted for NRC review in Section 11.3 of the Claiborne Enrichment Center Safety Analysis Report (LES, 1993). The NRC Staff reviewed the previous submittal and found it to be acceptable. The NRC staff's review and conclusions associated with Training are documented in Section 10.4 of NUREG-1491 (NRC, 1994).

11.3.1 Organization and Management of the Training Function

Line managers are responsible for the content and effective conduct of training for their personnel. Training responsibilities for line managers are included in position descriptions, and line managers are given the authority to implement training for their personnel. The training organization provides support to line managers by facilitating the planning, directing, analyzing, developing, conducting, evaluating, and controlling of a systematic performance-based training process. Performance-based training is used as the primary management tool for analyzing, designing, developing, conducting, and evaluating training.

Facility administrative procedures establish the requirements for indoctrination and training of personnel performing activities relied on for safety and to ensure that the training program is conducted in a reliable and consistent manner throughout all training areas. Exceptions from training requirements may be granted when justified and documented in accordance with procedures and approved by appropriate management.

Lesson plans are used for classroom and on-the-job training to provide consistent subject matter. When design changes or facility modifications are implemented, updates of applicable lesson plans are included in the change control process of the configuration management program.

Training records are maintained to support management information needs associated with personnel training, job performance, and qualifications.

The training programs at the facility are the responsibility of the Human Resources Manager. Records are maintained on each employee's qualifications, experience, training and retraining. The employee training file shall include records of all general employee training, technical training, and employee development training conducted at the facility. The employee training file shall also contain records of special company sponsored training conducted by others. The training records for each individual are maintained so that they are accurate and retrievable. Training records are retained in accordance with the records management procedures.

11.3.2 Analysis and Identification of Functional Areas Requiring Training

A needs/job analysis is performed and tasks are identified to ensure that appropriate training is provided to personnel working on tasks related to IROFS. Additionally, Job Hazard Analysis (JHA), sometimes referred to as Job Safety Analysis (JSA) (i.e., a step-by-step process used to evaluate job hazards), will be used as part of on-the-job training for providing employees the skills necessary to perform their jobs safely at the NEF.

The training organization consults with relevant technical and management personnel as necessary to develop a list of tasks for which personnel training for specific jobs is appropriate. The list of tasks selected for training is reviewed and compared to the training materials as part of the systematic evaluation of training effectiveness. The task list is also updated as necessitated by changes in procedures, processes, plant systems, equipment, or job scope.

11.3.3 Position Training Requirements

Minimum training requirements are developed for those positions whose activities are relied on for safety. Initial identification of job-specific training requirements is based on experience. Entry-level criteria (e.g., education, technical background, and/or experience) for these positions are contained in position descriptions.

The training program is designed to prepare initial and replacement personnel for safe, reliable and efficient operation of the facility. Appropriate training for personnel of various abilities and experience backgrounds is provided. The level at which an employee initially enters the training program is determined by an evaluation of the employee's past experience, level of ability, and qualifications.

Facility personnel may be trained through participation in prescribed parts of the training program that consists of the following:

- General Employee Training
- Technical Training
- Employee Development/Management-Supervisory Training.

Training is made available to facility personnel to initially develop and maintain minimum qualifications outlined in Chapter 2, Organization and Administration. The objective of the training shall be to ensure safe and efficient operation of the facility and compliance with applicable established regulations and requirements. Training requirements shall be applicable to, but not necessarily restricted to, those personnel within the plant organization who have a direct relationship to the operation, maintenance, testing or other technical aspect of the facility IROFS. Training courses are kept up-to-date to reflect plant modifications and changes to procedures when applicable.

Continuing or periodic retraining courses shall be established when applicable to ensure that personnel remain proficient. Periodic retraining generally is conducted to ensure retention of knowledge and skills important to facility operations. The training may consist of periodic retraining exercises, instruction, and review of subjects as appropriate to maintain proficiency of all personnel assigned to the facility. Section 7, Maintenance of Radiological Contingency Preparedness Capability, of the Emergency Plan provides additional information on personnel training for emergency response tasks.

11.3.3.1 General Employee Training

General Employee Training encompasses those Quality Assurance, radiation protection, safety, emergency and administrative procedures established by facility management and applicable regulations. The safety training for the NEF complies with the applicable sections of Occupational Safety and Health Administration (OSHA) regulations such as 29 CFR 1910 (Occupational Safety and Health Standards), 1910.1200 (Hazard Communication), and with NRC regulations such as 10 CFR 20 (Standards for Protection Against Radiation) and 10 CFR 19 (Notices, Instructions and Reports to Workers: Inspection and Investigations). Continuing training is conducted in these areas as necessary to maintain employee proficiency. All persons under the supervision of facility management (including contractors) must participate in General Employee Training; however, certain facility support personnel, depending on their normal work assignment, may not participate in all topics of this training. Temporary maintenance and service personnel receive General Employee Training to the extent necessary to assure safe execution of their duties. Certain portions of General Employee Training may be included in a New Employee Orientation Program.

General Employee Training topics are listed below:

- General administrative controls and procedure use
- Quality Assurance policies and procedures
- Facility systems and equipment
- Nuclear safety (See Section 11.3.3.1.1 - includes the use of dosimetry, protective clothing and equipment)
- Industrial safety, health and first aid
- Emergency Plan and implementing procedures
- Facility Security Programs (includes the protection of classified matter)
- Chemical Safety
- Fire Protection and Fire Brigade (see Section 11.3.3.1.2)
- New Employee Orientation.

11.3.3.1.1 Nuclear Safety Training

Training programs are established for the various types of job functions (e.g., production operator, radiation protection technician, contractor personnel) commensurate with criticality safety and/or radiation safety responsibilities associated with each such position. Visitors to the Controlled Access Area are trained in the formal training program or are escorted by trained personnel while in the Controlled Access Area.

This training is highlighted to stress the high level of importance placed on the radiological, criticality and chemical safety of plant personnel and the public. This training is structured as follows:

A. Personnel access procedures ensure the completion of formal nuclear safety training prior to permitting unescorted access into the Controlled Access Area.

B. Training sessions covering criticality safety, radiation protection and emergency procedures are conducted on a regular basis to accommodate new employees or those requiring retraining. Topics covered in the training program include:

- Notices, reports and instructions to workers
- Practices designed to keep radiation exposures ALARA
- Methods of controlling radiation exposures
- Contamination control methods (including decontamination)
- Use of monitoring equipment
- Emergency procedures and actions
- Nature and sources of radiation
- Safe use of chemicals
- Biological effects of radiation
- Use of personnel monitoring devices
- Principles of nuclear criticality safety
- Risk to pregnant females
- Radiation protection practices
- Protective clothing
- Respiratory protection
- Personnel surveys.

Criticality safety training shall be in accordance with ANSI/ANS-8.19-1996 (ANSI, 1996) and ANSI/ANS-8.20-1991 (ANSI, 1991).

Individuals attending these sessions must pass an initial examination covering the training contents to assure the understanding and effectiveness of the training. The effectiveness of the training programs is also evaluated by audits and assessments of operations and maintenance personnel responsible for following the requirements related to the topics listed above.

Newly hired or transferred employees reporting for work prior to the next regularly scheduled training session must complete nuclear safety training prior to unescorted access into the Controlled Access Area.

Since contractor employees perform diverse tasks in the Controlled Access Area, formal training for these employees is designed to address the type of work they perform. In addition to applicable radiation safety topics, training contents may include Radiation Work Permits, special bioassay sampling, and special precautions for welding, cutting, and grinding in the Controlled Access Area.

These training programs are conducted by instructors assigned by the HS&E Manager as having the necessary knowledge to address criticality safety and radiation protection. Records of the training programs are maintained as described in Section 11.7, "Records Management."

C. Individuals requiring unescorted access to the Controlled Access Area receive annual retraining. Retraining for individuals is scheduled and reported by means of a computerized tracking system.

D. Contents of the formal nuclear safety training programs are reviewed and updated periodically by the HS&E Manager, or designee, to ensure that the programs are current and adequate. In addition, at least annually, the contents of the radiation protection sections of the nuclear safety training program are reviewed and updated, as required, by the HS&E Manager or his designee.

E. Operational personnel are further instructed in the specific safety requirements of their work assignments by their immediate supervisor or delegate during on-the-job training. Employees must demonstrate understanding of work assignment requirements based on observations by their immediate supervisor or delegate before working without direct supervision. Changes to work procedures including safety requirements are reviewed with operational personnel by their immediate supervisor or delegate.

F. Radiation safety topics are also discussed and reviewed at least annually in roundtable safety meetings held by supervisors or delegates with their workers, and at other meetings held by managers with their employees.

11.3.3.1.2 Fire Brigade Training

The primary purpose of the Fire Brigade Training Program is to develop a group of facility employees skilled in fire prevention, fire fighting techniques, first aid procedures, and emergency response. They are trained and equipped to function as a team for the fighting of fires. The intent of the facility fire brigade is to be a first response effort designed to supplement the local fire department for fires at the plant and not to replace local fire fighters.

The Fire Brigade Training program provides for initial training of all new fire brigade members, semi-annual classroom training and drills, annual practical training, and leadership training for fire brigade leaders.

11.3.3.2 Technical Training

Technical training is designed, developed and implemented to assist facility employees in gaining an understanding of applicable fundamentals, procedures, and practices common to a gas centrifuge uranium enrichment facility. Also, technical training is used to develop manipulative skills necessary to perform assigned work in a competent manner. Technical training consists of four segments:

- Initial Training
- On-the-Job Training and Qualifications
- Continuing Training
- Special Training.

11.3.3.2.1 Initial Training

Initial job training is designed to provide an understanding of the fundamentals, basic principles, and procedures involved in work to which an employee is assigned. This training may consist of, but is not limited to, live lectures, taped and filmed lectures, self-guided study, demonstrations, laboratories and workshops and on-the-job training.

Certain new employees or employees transferred from other sections within the facility may be partially qualified by reason of previous applicable training or experience. The extent of further training for these employees is determined by applicable regulations, performance in review sessions, comprehensive examinations, or other techniques designed to identify the employee's present level of ability.

Initial job training and qualification programs are developed for operations, maintenance and technical services classifications. Training for each program is grouped into logical blocks or modules and presented in such a manner that specific behavioral objectives are accomplished. Trainee progress is evaluated using written examinations, oral or practical tests. Depending upon the regulatory requirements or individual's needs and plant operating conditions, allowances are made to suit specific situations. Brief descriptions of modules that may be contained in the initial training programs are as follows:

<u>Operations Initial Training</u>

A. General Systems

This training module provides the trainee with basic concepts and fundamentals in mathematics, physics, chemistry, heat transfer and electrical theory. Systems and components are taught in detail along with elementary process instrumentation and control. On-the-job orientation may be provided at an enrichment facility.

B. Specific Systems

This training module provides basic instruction in system and component identification and basic system operating characteristics. It provides a general overview of enrichment plant equipment and acquaints the trainees with enrichment plant terminology and nomenclature and provides instruction describing basic system operations.

C. Nuclear Preparatory

This training module develops the necessary concepts in basic nuclear physics, plant chemistry, basic thermodynamics, radiation protection, and enrichment theory. Experience in enrichment control and radiation protection is also provided. It is normally presented to operations personnel following the Systems Specific training module.

D. Plant Familiarization

The Plant Familiarization module provides for the orientation of employees to plant layout, plant systems, and practical laboratory and equipment work at the facility.

Mechanical Maintenance Initial Training

A. General Systems

This training module provides the trainee with basic concepts and fundamentals in mathematics, physics, chemistry, heat transfer and electrical theory. Systems and components are taught in detail along with elementary process instrumentation and control. On-the-job orientation may be provided at an enrichment facility.

B. Fundamental Shop Skills

This training module provides instruction in fundamentals of mechanical maintenance performance. It combines academic instruction with hands-on training to familiarize trainees with design operational and physical characteristics of enrichment facility components, and basic skills and procedures used to perform mechanical repairs and/or equipment replacement. Task training lists are integrated into this module to assure that each trainee attains a minimum level of performance. Tasks are assigned and trainees use work procedures to guide them through a task. Both radiological and industrial safety is stressed in all phases of this training module.

C. Plant Familiarization

The Plant Familiarization module provides for the orientation of employees to plant layout, plant systems, and practical laboratory and equipment work at the facility.

Instrumentation and Electrical and Maintenance Initial Training

A. General Systems

This training module provides the trainee with basic concepts and fundamentals in mathematics, physics, chemistry, heat transfer and electrical theory. Systems and components are taught in detail along with elementary process instrumentation and control. On-the-job orientation may be provided at an enrichment facility.

B. Basic Instrument and Electrical

This training module provides the trainee with refresher training in Electrical and Electronic Fundamentals, Digital Techniques and Application, Instrumentation and Control Theory and Application, and an introduction to the types and proper use of measuring and test equipment commonly used in enrichment facilities.

The module also provides the student a working knowledge of nuclear and non-nuclear instrumentation systems, overall integrated plant operation and control, and, in particular, the hazards of calibration errors and calibration during plant operation.

C. Basic Performance

The Fundamental Performance module familiarizes the trainee with plant test procedures, test equipment, and testing as well as plant records, reports, and data collection. It provides a basic understanding of thermodynamics used in testing plant heat transfer.

D. Plant Familiarization

The Plant Familiarization module provides for the orientation of employees to plant layout, plant systems, and practical laboratory and equipment work at the plant.

Health Physics and Chemistry Initial Training

A. General Systems

This training module provides the trainee with basic concepts and fundamentals in mathematics, physics, chemistry, heat transfer and electrical theory. Systems and components are taught in detail along with elementary process instrumentation and control. On-the-job orientation may be provided at an enrichment facility.

B. Fundamental Health Physics

The Fundamental Health Physics Module presents to the trainees a more comprehensive and theoretical understanding of the nuclear processes with which they are involved. In addition, the techniques for applying theory are presented in this module. Use is made of various non-automated counting and spectrographic equipment and portable survey instruments. Administrative material is also presented in a more detailed manner.

C. Fundamental Chemistry

The Fundamental Chemistry module provides familiarization with chemistry theory, techniques, and procedures. The overall goal of this module is familiarization necessary for chemistry technicians to be able to work safely and competently in the enrichment facility.

D. Plant Familiarization

The Plant Familiarization module provides for the orientation of employees to plant layout, plant systems, and practical laboratory and equipment work at the plant.

Engineer/Professional Initial Training

This training is part of the technical staff and managers training program.

A. Facility Orientation

This training module provides an orientation to each section within the NEF. An on-the-job task list provides the trainee with training objectives that must be accomplished while working in the section.

B. Basic Engineer/Professional Training

The Basic Engineer/Professional Training provides a basic understanding of how uranium is enriched, the systems and components required for producing the final product, and the interrelationship of the various facility organizations in achieving the overall objective.

C. Enrichment/Chemical Engineer/Professional Training

The Enrichment/Chemical Engineer/Professional Training provides specific theoretical information related to enrichment plant operations. Topics (e.g., Thermal Science, Nuclear Physics) address applications in an enrichment facility.

D. Engineer/Professional Systems Training

The Engineer/Professional Systems Training provides an overview of plant systems, components and procedures necessary to operate an enrichment plant safely and efficiently.

11.3.3.2.2 On-the-Job Training and Qualifications

On-the-job training (OJT) is a systematic method of providing the required job related skills and knowledge for a position. This training is conducted in the work environment. Applicable tasks and related procedures make up the OJT/qualifications program for each technical area which is designed to supplement and complement training received through formal classroom, laboratory, and/or simulator training. The objective of the program is to assure the trainee's ability to perform job tasks as described in the task descriptions and the Training and Qualification Guides.

11.3.3.2.3 Continuing Training

Continuing training is any training not provided as initial qualification and basic training which maintains and improves job-related knowledge and skills such as the following:

- Facility systems and component changes
- OJT/Qualifications program retraining
- Policy and procedure changes
- Operating experience program documents review to include Industry and in-house operating experiences
- Continuing training required by regulation (e.g., emergency plan training)
- General employee, special, administrative, vendor, and/or advanced training topics supporting tasks that are elective in nature

- Training identified to resolve deficiencies (task-based) or to reinforce seldom used knowledge skills

- Refresher training on initial training topics

- Structured pre-job instruction, mock-up training, and walk throughs

- Quality awareness.

Continuing Training and Retraining may overlap to some degree in definition; however, Retraining refers to specific training designed for proficiency maintenance.

Continuing Training consists of formal and informal components performed on a frequency needed to maintain proficiency on the job. Each Section's Continuing Training Program is developed from a systematic approach, using information from job performance and safe operation as a basis for determining the content of continuing training. Continuing training may be offered, as needed, on any of the topics listed above.

Once the objectives for Continuing Training have been established, the methods for conducting the training may vary. The method selected must provide clear evidence of objective accomplishment and consistency in delivery.

11.3.3.2.4 Special Training

Special training involves those subjects of a unique nature required for a particular area of work. Special training is usually given to selected personnel based on specific needs not directly related to disciplinary lines.

11.3.4 Basis and Objectives for Training

Learning objectives identify the training content, as established by needs/job analyses and position-specific requirements. The task list from the needs/job analysis is used to develop action statements that describe the desired post-training performance. Objectives include the knowledge, skills, and abilities the trainee should demonstrate; the conditions under which required actions will take place; and the standards of performance the trainee should achieve upon completion of the training activity.

11.3.5 Organization of Instruction, Using Lesson Plans and Other Training Guides

Lesson plans are developed from the learning objectives that are based on job performance requirements. Lesson plans and other training guides are developed under the guidance of the training function. Lesson plans are reviewed by the training function and, generally, by the organization cognizant of the subject matter. Lesson plans are approved prior to issue or use. Lesson plans are used for classroom training and on-the-job training as required and include Standards for evaluating acceptable trainee performance.

11.3.6 Evaluation of Trainee Learning

Trainee understanding and command of learning objectives is evaluated through observation/demonstration or oral or written tests as appropriate. Such evaluations measure the trainee's skills and knowledge of job performance requirements.

Evaluations are performed by individuals qualified in the training subject matter.

11.3.7 Conduct of On-the-Job Training

On-the-Job Training is an element of the technical training program (see Section 11.3.3.2.2, On-the-Job Training and Qualifications). On-the-job training is used in combination with classroom training for activities that are IROFS. Designated personnel who are competent in the program standards and methods of conducting the training conduct on-the-job training using current performance-based training materials. Completion of on-the-job training is demonstrated by actual task performance or performance of a simulation of the task with the trainee explaining task actions using the conditions encountered during the performance of the task, including references, tools, and equipment reflecting the actual task to the extent practical.

11.3.8 Evaluation of Training Effectiveness

Periodically the training program is systematically evaluated to measure the program's effectiveness in producing competent employees. The trainees provide feedback after completion of classroom training sessions to provide data for this evaluation for program improvements. These evaluations identify program strengths and weaknesses, determine whether the program content matches current job needs, and determine if corrective actions are needed to improve the program's effectiveness. The training function is responsible for leading the training program evaluations and for implementing any corrective actions. Program evaluations may consist of an overall periodic evaluation or a series of topical evaluations over a given period.

Evaluation objectives that are applicable to the training program or topical area being reviewed are developed and may address the following elements of training:

- Management and administration of training and qualification programs
- Development and qualification of the training staff
- Position training requirements
- Determination of training program content, including its facility change control interface with the configuration management system
- Design and development of training programs, including lesson plans
- Conduct of training
- Trainee examinations and evaluations

- Training program assessments and evaluations.

Evaluation results are documented, with program strengths and weaknesses being highlighted. Identified weaknesses are reviewed, improvements are recommended, and changes are made to procedures, practices, or training materials as necessary.

Periodically, training and qualifications activities are monitored by designated facility and/or contracted training personnel. The Quality Assurance Department audits the facility training and qualification system. In addition, trainees and vendors may provide input concerning training program effectiveness. Methods utilized to obtain this information include, among other things surveys, questionnaires, performance appraisals, staff evaluation, and overall training program effectiveness evaluation instruments. Frequently conducted classes are not evaluated each time. However, they are routinely evaluated at a frequency sufficient to determine program effectiveness. Evaluation information may be collected through:

- Verification of program objectives as related to job duties for which intended
- Periodic working group program evaluations
- Testing to determine trainee accomplishment of objectives
- Trainee evaluation of the instruction
- Supervisor's evaluation of the trainee's performance after training on-the-job
- Supervisor's evaluation of the instruction.

Unacceptable individual performance is transmitted to the appropriate Line Manager.

11.3.9 Personnel Qualification

The qualification requirements for key management positions are described in Chapter 2, Organization and Administration. Training and qualification requirements associated with QA personnel are provided in Appendix A to this chapter. In addition, qualification and training requirements for process operator candidates shall be established and implemented in plant procedures.

11.3.10 Periodic Personnel Evaluations

Personnel performing activities relied on for safety are evaluated at least biennially to determine whether they are capable of continuing their activities that are relied on for safety. The evaluation may be by written test, oral test, or on-the-job performance evaluation. The results of the evaluation are documented. When the results of the evaluation dictate, retraining or other appropriate action is provided. Retraining is also required due to plant modifications, procedure changes, and QA program changes that result in new or revised information.

11.4 PROCEDURES DEVELOPMENT AND IMPLEMENTATION

The management measure described in this section is consistent with that previously submitted for NRC review in Section 11.4 of the Claiborne Enrichment Center Safety Analysis Report (LES, 1993). The NRC staff reviewed the previous submittal and found it to be acceptable. The NRC staff's review and conclusions associated with procedures are documented in Section 10.5 of NUREG-1491 (NRC, 1994).

The requirements for independent verification are consistent with the applicable guidance provided in ANSI/ANS-3.2-1994, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants."

All activities involving licensed materials or IROFS are conducted in accordance with approved procedures. Before initial enrichment activities occur at the facility, procedures are made available to the NRC for their inspection. As noted throughout this document, procedures are used to control activities in order to ensure the activities are carried out in a safe manner and in accordance with regulatory requirements.

Generally, four types of plant procedures are used to control activities: operating procedures, administrative procedures, maintenance procedures, and emergency procedures.

Operating procedures, developed for workstation and Control Room operators, are used to directly control process operations. Operating procedures include:

- Purpose of the activity
- Regulations, polices, and guidelines governing the procedure
- Type of procedure
- Steps for each operating process phase:
  - Initial startup
  - Normal operations
  - Temporary operations
  - Emergency shutdown
  - Emergency operations
  - Normal shutdown
  - Startup following an emergency or extended downtime.
- Hazards and safety considerations
- Operating limits
- Precautions necessary to prevent exposure to hazardous chemicals (resulting from operations with Special Nuclear Material (SNM)) or to licensed SNM.

- Measures to be taken if contact or exposure occurs
- IROFS associated with the process and their functions
- The timeframe for which the procedure is valid.

Applicable safety limits and IROFS are clearly identified in the procedures. LES will incorporate methodology for identifying, developing, approving, implementing, and controlling operating procedures. Identifying needed procedures will include consideration of ISA results. The method will ensure that, as a minimum:

- Operating limits and IROFS are specified in the procedure
- Procedures include required actions for off-normal conditions of operation, as well as normal operations
- If needed safety checkpoints are identified at appropriate steps in the procedure
- Procedures are validated through field tests
- Procedures are approved by management personnel responsible and accountable for the operation
- A mechanism is specified for revising and reissuing procedures in a controlled manner
- The QA elements and CM Program at the facility provide reasonable assurance that current procedures are available and used at all work locations
- The facility training program trains the required persons in the use of the latest procedures available.

Administrative procedures are used to perform activities that support the process operations, including management measures such as the following:

- Configuration management
- Nuclear criticality, radiation, chemical, and fire safety
- Quality Assurance
- Design control
- Plant personnel training and qualification
- Audits and assessments
- Incident investigations
- Record keeping and document control

- Reporting
- Procurement.

Administrative procedures are also used for:

- Implementing the Fundamental Nuclear Material Control (FNMC) Plan
- Implementing the Emergency Plan
- Implementing the Physical Security Plan
- Implementing the Standard Practice Procedures Plan for the Protection of Classified Matter.

Maintenance procedures address:

- Preventive and corrective maintenance of IROFS
- Surveillance (includes calibration, inspection, and other surveillance testing)
- Functional testing of IROFS
- Requirements for pre-maintenance activity involving reviews of the work to be performed and reviews of procedures.

Emergency procedures address the preplanned actions of operators and other plant personnel in the event of an emergency.

Procedures will be established and implemented for nuclear criticality safety in accordance with ANSI/ANS-8.19-1996 (ANSI, 1996). The NCS procedures will be written such that no single, inadvertent departure from a procedure could cause an inadvertent criticality. Nuclear criticality safety postings at the NEF are established that identify administrative controls applicable and appropriate to the activity or area in question. Nuclear criticality safety procedures and postings are controlled by procedure to ensure that they are maintained current.

Periodic reviews will be performed on procedures to assure their continued accuracy and usefulness. Specifically, reviews of operating procedures will be conducted at a minimum of every five years and reviews of radiation protection procedures and emergency procedures will be conducted at a minimum of every year. In addition, applicable procedures will be reviewed after unusual incidents, such as an accident, unexpected transient, significant operator error, or equipment malfunction, or after any modification to a system, and procedures will be revised as needed.

11.4.1 Preparation of Procedures

Each procedure is assigned to a member of the facility staff or contractor for development. Initial procedure drafts are reviewed by other appropriate members of the facility staff, by personnel from the supplier of centrifuges (Urenco), and other vendors, as appropriate for inclusion and correctness of technical information, including formulas, set points, and acceptance criteria and includes either a walkdown of the procedure in the field or a tabletop walkthrough. Procedures that are written for the operation of IROFS shall be subjected to an independent review. The designated approver shall determine whether or not any additional, cross-disciplinary review is required. The Plant Manager or designee shall approve all procedures. If the procedure involves QA directly, the QA Manager must approve the procedure.

11.4.2 Administrative Procedures

Facility administrative procedures are written by each department as necessary to control activities that support process operations, including management measures. Listed below are several areas for which administrative procedures are written, including principle features:

A. Operator's authority and responsibility: The operator is given the authority to manipulate controls which directly or indirectly affect the enrichment process, including a shut down of the process if deemed necessary by the Shift Manager. The operators are also assigned the responsibility for knowing the limits and set points associated with safety-related equipment and systems as specified in designated operating procedures.

B. Activities affecting facility operation or operating indications: All facility maintenance personnel performing support functions (e.g., maintenance, testing) which may affect unit operation or Control Room indications are required to notify the Control Room Operator and/or Shift Manager, as appropriate, prior to initiating such action.

C. Manipulation of facility control: No one is permitted to manipulate the facility controls who is not an operator, except for operator trainees under the direction of a qualified operator.

D. Relief of Duties: This procedure provides a detailed checklist of applicable items for shift turnover.

E. Equipment control: Equipment control is maintained and documented through the use of tags, labels, stamps, status logs or other suitable means.

F. Master surveillance testing schedule: A master surveillance testing schedule is documented to ensure that required testing is performed and evaluated on a timely basis. Surveillance testing is scheduled such that the safety of the facility is not dependent on the performance of a structure, system or component which has not been tested within its specified testing interval. The master surveillance testing schedule identifies surveillance and testing requirements, applicable procedures, and required test frequency. Assignment of responsibility for these requirements is also indicated.

G. A Control Room Operations Logbook is maintained. This logbook contains significant events during each shift such as enrichment changes, alarms received, or abnormal operational conditions.

H. Fire Protection Procedures: Fire protection procedures are written to address such topics as training of the fire brigade, reporting of fires, and control of fire stops. The facility's Industrial Safety department has responsibility for fire protection procedures in general, with the facility's maintenance section having responsibility for certain fire protection procedures such as control of repairs to facility fire stops.

The administrative control of maintenance is maintained as follows:

A. In order to assure safe, reliable, and efficient operation, a comprehensive maintenance program for the facility's IROFS is established.

B. Personnel performing maintenance activities are qualified in accordance with applicable codes and standards and procedures.

C. Maintenance is performed in accordance with written procedures that conform to applicable codes, standards, specifications, and other appropriate criteria.

D. Maintenance is scheduled so as not to jeopardize facility operation or the safety of facility personnel.

E. Maintenance histories are maintained on facility IROFS.

The administrative control of facility modifications is discussed in Section 2.3.1, Configuration Management.

11.4.3 Procedures

All activities involving licensed materials or IROFS are conducted in accordance with approved procedures. These procedures are intended to provide a pre-planned method of conducting operations of systems in order to eliminate errors due to on-the-spot analysis and judgments.

All procedures are sufficiently detailed that qualified individuals can perform the required functions without direct supervision. However, written procedures cannot address all contingencies and operating conditions. Therefore, they contain a degree of flexibility appropriate to the activities being performed. Procedural guidance exists to identify the manner in which procedures are to be implemented. For example, routine procedural actions may not require the procedure to be present during implementation of the actions, while complex jobs, or checking with numerous sequences may require valve alignment checks, approved operator aids, or in-hand procedures that are referenced directly when the job is conducted.

Examples of operating activities are:

- Evacuation and Preparatory Work Before Run Up of a Cascade
- Run Up of a Cascade
- Run Down of a Cascade
- Calibration of Pressure Transmitter
- Taking UF_6 Samples of a Cascade
- Installation of UF_6 Cylinders in Feed/Take-off Stations and Preparation for Operation
- Removal of UF_6 Cylinder from Feed/Take-off Stations
- Installation of UF_6 Cylinders in Take-off Stations
- UF_6 Gas Sampling in Take-off Lines
- UF_6 Sampling in Product Liquid Sampling Autoclaves

- Emptying of Cold Trap
- Exchange of Chemical Traps in Vent Systems.

Plant specific procedures for abnormal events are written for the facility. These procedures are based on a sequence of observations and actions, with emphasis placed on operator responses to indications in the Control Room. When immediate operator actions are required to prevent or mitigate the consequences of an abnormal situation, procedures require that those actions be implemented at the earliest possible time, even if full knowledge of the abnormal situation is not yet available. The actions outlined in abnormal event procedures are based on a conservative course of action to be followed by the operating crew.

Typical abnormal event procedures include:

- Power Failure
- Loss of Heat Tracing
- Damaged UF_6 Cylinder Repairs
- Annunciator alarms (procedures to include alarm set points, probable causes, automatic actions, immediate manual actions, supplementary actions and applicable references).

Temporary changes to procedures are issued for operating activities that are of a nonrecurring nature. Temporary changes to procedures are used when revision of an operating or other permanent procedure is not practical. Temporary changes to procedures shall not involve a change to the ISA and shall not alter the intent of the original procedure. Examples of uses of temporary changes to procedures are:

- To direct operating activities during special testing or maintenance
- To provide guidance in unusual situations not within the scope of normal procedures
- To ensure orderly and uniform operations for short periods of time when the facility, a unit, a cascade, a structure, a system or a component is performing in a manner not addressed by existing procedures or has been modified in such a manner that portions of existing procedures do not apply.

The temporary changes to procedures are approved by two members of the facility management staff, at least one of whom is a shift manager. Temporary changes to procedures are documented, reviewed and approved with the process described in Section 11.4.4, Changes to Procedures, within 14 days of implementation.

Maintenance of facility structures, systems and components is performed in accordance with written procedures, documented instructions, checklists, or drawings appropriate to the circumstances (for example, skills normally possessed by qualified maintenance personnel may not require detailed step-by-step delineation in a written procedure) that conform to applicable codes, standards, specifications, and other appropriate criteria.

The facility's maintenance department under the Maintenance Manager has responsibility for preparation and implementation of maintenance procedures. The maintenance, testing and calibration of facility IROFS is performed in accordance with approved written procedures.

Testing conducted on a periodic basis to determine various facility parameters and to verify the continuing capability of IROFS to meet performance requirements is conducted in accordance with approved, written procedures. Periodic test procedures are utilized to perform such testing and are sufficiently detailed that qualified personnel can perform the required functions without direct supervision. Testing performed on IROFS that are not redundant will provide for compensatory measures to be put into place to ensure that the IROFS performs until it is put back into service.

Periodic test procedures are performed by the facility's Technical Services, Operations and Maintenance departments. The Maintenance Manager has overall responsibility for assuring that the periodic testing is in compliance with the requirements.

Chemical and radiochemical activities associated with facility IROFS are performed in accordance with approved, written procedures. The facility's chemistry department has responsibility for preparation and implementation of chemistry procedures.

Radioactive waste management activities associated with the facility's liquid, gaseous, and solid waste systems are performed in accordance with approved written procedures. The facility's operations, chemistry and radiation protection departments have responsibility for preparation and implementation of the radioactive waste management procedures.

Likewise, other departments at the facility develop and implement activities at the facility through the use of procedures.

Procedures will include provisions for operations to stop and place the process in a safe condition if a step of a procedure cannot be performed as written.

11.4.4 Changes to Procedures

Changes to procedures shall be processed as described below.

A. The preparer documents the change as well as the reason for the change.

B. An evaluation shall be performed in accordance with 10 CFR 70.72 (CFR, 2003e) as appropriate. If the evaluation reveals that a change to the license is needed to implement the proposed changes, the change is not implemented until prior approval is received from the NRC.

C. The procedure with proposed changes shall be reviewed by a qualified reviewer.

D. The Plant Manager, a Department Manager, or a designee approved by the Plant Manager shall be responsible for approving procedure changes, and for determining whether a cross-disciplinary review is necessary, and by which department(s). The need for the following cross-disciplinary reviews shall be considered, as a minimum:

   1. For proposed changes having a potential impact on chemical or radiation safety, a review shall be performed for chemical and radiation hazards. Changes shall be approved by the HS&E Manager or designee.

2. For proposed changes having a potential impact on criticality safety, an NCS evaluation and, if required, an NCS analysis shall be performed. Any necessary controlled parameters, limits, IROFS, management measures, or NCS analyses that must be imposed or revised are adequately reflected in appropriate procedures and/or design basis documents. Changes shall be independently reviewed by a criticality safety engineer, and approved by the HS&E Manager or designee.

3. For proposed changes potentially affecting Material Control and Accounting, a material control review shall be performed. Changes shall be approved by the HS&E Manager or designee.

Records of completed cross-functional reviews shall be maintained in accordance with Section 11.7, Records Management, for all changes to procedures involving licensed materials or IROFS.

11.4.5 Distribution of Procedures

Originally issued approved procedures and approved procedure revisions are distributed in a controlled manner by document control.

Document Control shall establish and maintain an index of the distribution of copies of all facility procedures. Revisions are controlled and distributed in accordance with this index. Indexes are reviewed and updated on a periodic basis or as required.

Department Managers or their designees shall be responsible for ensuring all personnel doing work which require the use of the procedures have ready access to controlled copies of the procedures.

11.5 AUDITS AND ASSESSMENTS

LES will have a tiered approach to verifying compliance to procedures and performance to regulatory requirements. Audits are focused on verifying compliance with regulatory and procedural requirements and licensing commitments. Assessments are focused on effectiveness of activities and ensuring that IROFS, and any items that affect the function of IROFS, are reliable and are available to perform their intended safety functions. This approach includes performing Assessments and Audits on critical work activities associated with facility safety, environmental protection and other areas as identified via trends.

Assessments are divided into two categories that will be owned and managed by the line organizations as follows:

- Management Assessments conducted by the line organizations responsible for the work activity
- Independent Assessments conducted by individuals not involved in the area being assessed.

Audits of the QA Level 1 work activities associated with IROFS and any items that affect the function of the IROFS and items required to satisfy regulatory requirements for which QA Level 1 requirements are applied will be the responsibility of the QA Department.

Audits and assessments are performed to assure that facility activities are conducted in accordance with the written procedures and that the processes reviewed are effective. As a minimum, they shall assess activities related to radiation protection, criticality safety control, hazardous chemical safety, industrial safety including fire protection, and environmental protection.

Audits and assessments shall be performed routinely by qualified staff personnel that are not directly responsible for production activities. Deficiencies identified during the audit or assessment requiring corrective action shall be forwarded to the responsible manager of the applicable area or function for action in accordance with the CAP procedure. Future audits and assessments shall include a review to evaluate if corrective actions have been effective.

The Quality Assurance Department shall be responsible for audits. Audits shall be performed in accordance with a written plan that identifies and schedules audits to be performed. Audit team members shall not have direct responsibility for the function and area being audited. Team members shall have technical expertise or experience in the area being audited and shall be indoctrinated in audit techniques. Audits shall be conducted on an annual basis.

The results of the audits shall be provided in a written report in a timely manner to the Plant Manager, the Safety Review Committee (SRC), and the Managers responsible for the activities audited. Any deficiencies noted in the audits shall be responded to promptly by the responsible Managers or designees, entered into the CAP and tracked to completion and re-examined during future audits to ensure corrective action has been completed.

Records of the instructions and procedures, persons conducting the audits or assessments, and identified violations of license conditions and corrective actions taken shall be maintained.

The management measure described in this section and Chapter 2, Organization and Administration, is consistent with that previously submitted for NRC review in the Claiborne

Enrichment Center Safety Analysis Report (LES, 1993). The NRC Staff reviewed the previous submittal and found it to be acceptable. The NRC Staff's review and conclusions associated with audits and assessments are documented in Section 10.7 of NUREG-1491 (NRC, 1994).

11.5.1 Activities to be Audited or Assessed

Audits and assessments are conducted for the areas of:

- Radiation safety
- Nuclear criticality safety
- Chemical safety
- Industrial safety including fire protection
- Environmental protection
- Emergency management
- QA
- Configuration management
- Maintenance
- Training and qualification
- Procedures
- CAP/Incident investigation
- Records management.

Assessments of nuclear criticality safety, performed in accordance with ANSI/ANS-8.19-1996 (ANSI, 1996), will ensure that operations conform to criticality requirements.

11.5.2 Scheduling of Audits and Assessments

A schedule is established that identifies audits and assessments to be performed and the responsible organization assigned to conduct the activity. The frequency of audits and assessments is based upon the status and safety importance of the activities being performed and upon work history. All major activities will be audited or assessed on an annual basis. The audit and assessment schedule is reviewed periodically and revised as necessary to ensure coverage commensurate with current and planned activities.

Nuclear Criticality safety audits are conducted and documented quarterly such that all aspects of the Nuclear Criticality Safety Program will be audited at least every two years. The Operations Group is assessed periodically to ensure that nuclear critical safety procedures are being followed and the process conditions have not been altered to adversely affect nuclear criticality safety. The frequency of these assessments is based on the controls identified in the NCS analyses and NCS evaluations. Assessments are conducted at least semi-annually. In addition, weekly nuclear criticality safety walkthroughs of UF_6 process areas are conducted and documented.

11.5.3 Procedures for Audits and Assessments

Internal and external audits and assessments are conducted using approved procedures that meet the QA Program requirements. These procedures provide requirements for the following audit and assessment activities:

- Scheduling and planning of the audit and assessment
- Certification requirements of audit personnel
- Development of audit plans and audit and assessment checklists as applicable
- Performance of the audit and assessment
- Reporting and tracking of findings to closure
- Closure of the audit and assessment.

The applicable procedures emphasize reporting and correction of findings to prevent recurrence.

Audits and assessments are conducted by:

- Using the approved audit and assessment checklists as applicable
- Interviewing responsible personnel
- Performing plant area walkdowns
- Reviewing controlling plans and procedures
- Observing work in progress
- Reviewing completed QA documentation.

Audit and assessment results are tracked in the Corrective Action Program. The data is periodically analyzed for potential trends and needed program improvements to prevent recurrence and/or for continuous program improvements. The resulting trend is evaluated and reported to applicable management. This report documents the effectiveness of management measures in controlling activities, as well as deficiencies. Deficiencies identified in the trend report require corrective action in accordance with the applicable CAP procedure. The QA organization also performs follow up reviews on identified deficiencies and verifies completion of corrective actions reported as a result of the trend analysis.

The audit and /or assessment team leader is required to develop the audit and /or assessment report documenting the findings, observations, and recommendations for program improvement. These reports provide management with documented verification of performance against established performance criteria for IROFS. These reports are developed, reviewed, approved, and issued following established formats and protocols detailed in the applicable procedures. Responsible managers are required to review the reports and provide any required responses due to reported findings.

Corrective actions following issuance of the audit and/or assessment report require compliance with the CAP procedure. Audit reports are required to contain an effectiveness evaluation and statement for each of the applicable QA program elements reviewed during the audit. The audit/assessment is closed with the proper documentation as required by the applicable audit and assessment procedure. The QA organization will conduct follow-up audits or assessments to verify that corrective actions were taken in a timely manner. In addition, future assessments will include a review to evaluate if corrective actions have been effective.

11.5.4 Qualifications and Responsibilities for Audits and Assessments

The QA Director or QA Manager initiates audits. The responsible Lead Auditor and QA Director or Manager determines the scope of each audit. The QA Director or QA Manager may initiate special audits or expand the scope of audits. The Lead Auditor directs the audit team in developing checklists, instructions, or plans and performing the audit. The audit shall be conducted in accordance with the checklists, but the scope may be expanded by the audit team during the audit. The audit team consists of one or more auditors.

Auditors and lead auditors are responsible for performing audits in accordance with the applicable QA procedures. Auditors and lead auditors hold certifications as required by the QA Program. Additional details can be found in Appendix A of this chapter. Before being certified under the LES QA Program, auditors must complete training on the following topics:

- LES QA Program
- Audit fundamentals, including audit scheduling, planning, performance, reporting, and follow-up action involved in conducting audits
- Objectives and techniques of performing audits
- On-the-job training.

Certification of auditors and lead auditors is based on the QA Director's or Manager's evaluation of education, experience, professional qualifications, leadership, sound judgment, maturity, analytical ability, tenacity, and past performance and completion of QA training courses. A lead auditor must also have participated in a minimum of five QA audits or audit equivalent within a period of time not to exceed three years prior to the date of certification. Audit equivalents include assessments, pre-award evaluations or comprehensive surveillances (provided the prospective lead auditor took part in the planning, checklist development, performance, and reporting of the audit equivalent activities). One audit must be a nuclear-related QA audit or audit equivalent within the year prior to certification.

Personnel performing assessments do not require certification, but they are required to complete QA orientation training, as well as training on the assessment process. The nuclear criticality safety assessments are performed under the direction of the criticality safety staff. Personnel performing these assessments do not report to the production organization and have no direct responsibility for the function or area being assessed.

Appendix A, Section 18 "Audits" of this chapter provides additional details regarding the QA Audit program requirements.

11.6 INCIDENT INVESTIGATIONS AND CORRECTIVE ACTION PROCESS

The incident investigation and corrective action process described in this section is consistent with that previously submitted for NRC review in Section 11.4 and Section 10.16 of the Claiborne Enrichment Center Safety Analysis Report (LES, 1993). The NRC Staff reviewed the previous submittal and found it to be acceptable. The NRC Staff's review and conclusions associated with the incident investigation and corrective action process are documented in Section 10.7.6 and Section 12 of NUREG-1491 (NRC, 1994).

11.6.1 Incident Investigations

The incident investigation process is a simple mechanism available for use by any person at the facility for reporting deficiencies, abnormal events and potentially unsafe conditions or activities. Abnormal events that potentially threaten or lessen the effectiveness of health, safety or environmental protection will be identified and reported to and investigated by the HS&E Manager. Each event will be considered in terms of its requirements for reporting in accordance with regulations and will be evaluated to determine the level of investigation required. The process of incident identification, investigation, root cause analysis, environmental protection analysis, recording, reporting, and follow-up shall be addressed in and performed by written CAP procedures. Radiological, criticality, hazardous chemical, and industrial safety requirements shall be addressed. Guidance for classifying occurrences shall be contained in CAP procedures, including examples of threshold off-normal occurrences. The depth of the investigation will depend upon the severity of the classified incident in terms of the levels of uranium released and/or the degree of potential for exposure of workers, the public or the environment.

The HS&E Manager is responsible for:

- Maintaining a list of agencies to be notified
- Determining if a report to an agency is required
- Notifying the agency when required.

The licensing organization has the responsibility for all appropriate communications with government agencies.

The HS&E Manager or designee shall maintain a record of corrective actions to be implemented as a result of off-normal occurrence investigations in accordance with CAP procedures. These corrective actions shall include documenting lessons learned, and implementing worker training where indicated, and shall be tracked to completion by the HS&E Manager or designee.

Specifics of the Incident Investigation process are as follows:

1. LES will establish a process to investigate abnormal events that may occur during operation of the facility, to determine their specific or generic root cause(s)and generic implications, to recommend corrective actions, and to report to the NRC as required by 10 CFR 70.50 (CFR, 2003c) and 70.74 (CFR, 2003f). The investigation process will include a prompt risk-based evaluation and, depending on the complexity and severity of

the event, one individual may suffice to conduct the evaluation. The investigator(s) will be independent from the line function(s) involved with the incident under investigation and are assured of no retaliation for participating in investigations. Investigations will begin within 48 hours of the abnormal event, or sooner, depending on safety significance of the event. The record of IROFS failures required by 10 CFR 70.62(a)(3) (CFR, 2003d) for IROFS will be reviewed as part of the investigation. Record revisions necessitated by post-failure investigation conclusions will be made within five working days of the completion of the investigation.

2. Qualified internal or external investigators are appointed to serve on investigating teams when required. The teams will include at least one process expert and at least one team member trained in root cause analysis.
3. LES will monitor and document corrective actions through completion.
4. LES will maintain auditable records and documentation related to abnormal events, investigations, and root cause analyses so that "lessons learned" may be applied to future operations of the facility. For each abnormal event, the incident report includes a description, contributing factors, a root cause analysis, findings, and recommendations. Relevant findings are reviewed with all affected personnel. Details of the event sequence will be compared with accident sequences already considered in the ISA, and the ISA Summary will be modified to include evaluation of the risk associated with accidents of the type actually experienced.

LES will develop CAP procedures for conducting an incident investigation, and the procedures will contain the following elements:

1. A documented plan for investigating an abnormal event.
2. A description of the functions, qualifications, and/or responsibilities of the manager who would lead the investigative team and those of the other team members; the scope of the team's authority and responsibilities; and assurance of cooperation of management.
3. Assurance of the team's authority to obtain all the information considered necessary and its independence from responsibility for or to the functional area involved in the incident under investigation.
4. Retention of documentation relating to abnormal events for two years or for the life of the operation, whichever is longer.
5. Guidance for personnel conducting the investigation on how to apply a reasonable, systematic, structured approach to determine the specific or generic root cause(s) and generic implications of the problem.
6. Requirements to make available original investigation reports to the NRC on request.
7. A system for monitoring the completion of appropriate corrective actions.

11.6.2 Corrective Action Process

The LES QA Program identifies the responsibilities and provides authority for those individuals involved in quality activities to identify any condition adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective materials and equipment, and nonconformances. These individuals identify and document conditions adverse to quality, analyze and determine how the conditions can be corrected or resolved, and take such steps as necessary to implement corrective actions in accordance with documented procedures.

The QA Program requires regularly scheduled audits and assessments to ensure that needed corrective actions are identified. LES employees have the authority and responsibility to initiate the corrective action process if they discover deficiencies. The QA Program contains procedures for identifying, reporting, resolving, documenting, and analyzing conditions adverse to quality. Reports of conditions adverse to quality are analyzed to identify trends in quality performance. Significant conditions adverse to quality and significant trends are reported to senior management in accordance with CAP procedures.

Follow-up action is taken by the QA Manager to verify proper and timely implementation of corrective action.

Significant conditions adverse to quality, the cause of the conditions and the corrective action taken to preclude repetition are documented and reported to management for review and assessment in accordance with CAP procedures.

Appendix A, Section 16 “Corrective Action” of this chapter provides additional details regarding the CAP requirements.

11.7 RECORDS MANAGEMENT

The management measure described in this section is consistent with that previously submitted for NRC review in Section 11.4 of the Claiborne Enrichment Center Safety Analysis Report (LES, 1993). The NRC Staff reviewed the previous submittal and found it to be acceptable. The NRC Staff's review and conclusions associated with records management are documented in Section 10.6 of NUREG-1491 (NRC, 1994).

Records management shall be performed in a controlled and systematic manner in order to provide identifiable and retrievable documentation. Applicable design specifications, procurement documents, or other documents specify the QA records to be generated by, supplied to, or held, in accordance with approved procedures. QA records are not considered valid until they are authenticated and dated by authorized personnel.

The LES QA Program requires procedures for reviewing, approving, handling, identifying, retention, retrieval and maintenance of quality assurance records. These records include the results of tests and inspections required by applicable codes and standards, construction, procurement and receiving records, personnel certification records, design calculations, purchase orders, specifications and amendments, procedures, incident investigation results and approvals or corrective action taken, various certification forms, source surveillance and audit reports, component data packages, and any other QA documentation required by specifications or procedures. These records are maintained at locations where they can be reviewed and audited to establish that the required quality has been assured.

For computer codes and computerized data used for activities relied on for safety, as specified in the ISA Summary, procedures are established for maintaining readability and usability of older codes and data as computing technology changes. For example, procedures allow older forms of information and codes for older computing equipment to be transferred to contemporary computing media and equipment.

The facility maintains a Master File that access to, and use of is controlled. Documents in the Master File shall be legible and shall be identifiable as to the subject to which they pertain. Documents shall be considered valid only if stamped, initialed, signed or otherwise authenticated and dated by authorized personnel. Documents in the Master File may be originals or reproduced copies. Computer storage of data may be used in the Master File.

In order to preclude deterioration of records in the Master File, the following requirements are applicable:

A. Records shall not be stored loosely. Records shall be firmly attached in binders or placed in folders or envelopes. Records should be stored in steel file cabinets.

B. Special processed records, e.g., radiographs, photographs, negatives, microfilm, which are light-sensitive, pressure-sensitive and/or temperature-sensitive, shall be packaged and stored as recommended by the manufacturer of these materials.

C. Computer storage of records shall be done in a manner to preclude inadvertent loss and to ensure accurate and timely retrieval of data. Dual-facility records storage uses an electronic data management system and storage of backup tapes in a fireproof safe.

The Master File storage system shall provide for the accurate retrieval of information without undue delay. Written instructions shall be prepared regarding the storage of records in a Master

File, and a supervisor shall be designated the responsibility for implementing the requirements of the instructions. These instructions shall include, but not necessarily be limited to the following.

A. A description of the location(s) of the Master File and an identification of the location(s) of the various record types within the Master File

B. The filing system to be used

C. A method for verifying that records received are in agreement with any applicable transmittal documents and are in good condition. This is not required for documents generated within a section for use and storage in the same sections' satellite files.

D. A method for maintaining a record of the records received

E. The criteria governing access to and control of the Master File

F. A method for maintaining control of and accountability for records removed from the Master File

G. A method for filing supplemental information and for disposing of superseded records.

A qualified Fire Protection Engineer will evaluate record storage areas (including satellite files) to assure records are adequately protected from damage.

Records related to health and safety shall be maintained in accordance with the requirements of Title 10, Code of Federal Regulations. The following records shall be retained for at least the periods indicated in accordance with the Records Management procedures which specifies retention periods

The following are examples of records that will be retained:

- Operating logs
- Procedures
- Supplier QA documentation for equipment, materials, etc.
- Nonconforming item reports
- Test documentation/test results - preoperational/operational
- Facility modification records
- Drawings/specifications
- Procurement documents (e.g., purchase orders, purchase requisitions)
- Nuclear material control and accounting records
- Maintenance activities including calibration records
- Inspection documentation (plant processes)

- Audit reports
- Reportable occurrences and compliance records
- Completed work orders
- License conditions (specifications) records
- Software verification records
- System descriptions
- As-built design documentation packages
- Regulatory reports and corrective action.

Other retention times are specified for other facility records as necessary to meet applicable regulatory requirements. These retention times are indicated in facility administrative procedures.

Appendix A, Section 17 “Quality Assurance Records” of this chapter provides additional details regarding records management requirements.

11.8 OTHER QA ELEMENTS

The QA Program and its supporting manuals, procedures and instructions are applicable to items and activities designated as QA Level 1 and 2.

The QA Director is responsible for developing and revising the QA Program and assuring it is in compliance with applicable regulations, codes and standards. The QA Director approves the supporting manuals, procedures, and revisions for their respective scope of responsibility.

The QA Program specifies mandatory requirements for performing activities affecting quality and is set forth in procedures which are distributed on a controlled basis to organizations and individuals responsible for quality. Revisions to these procedures are also distributed on a controlled basis. Applicable portions of the QA Program are documented, approved and implemented prior to undertaking an activity.

A management assessment of the QA program is performed at least six months prior to scheduled receipt of licensed material on the site. Items identified as needing completion or modification are entered into the CAP and corrective action completed before scheduled receipt of licensed material. LES Management monitors the QA program prior to this initial management assessment through project review meetings and annual assessments. This management assessment along with integrated schedules and program review meetings ensure that the QA program is in place and effective prior to receiving licensed material.

The LES QA program for design, construction, and preoperational testing continues simultaneously with the QA program for the operational phase while construction activities are in progress.

Anyone may propose changes to the QA Program supporting manuals and procedures. When reviewed by the QA Director and found acceptable and compatible with applicable requirements, guidelines and LES policy, the changes may be implemented. The QA Program and supporting manuals and procedures are reviewed periodically to ensure they are in compliance with applicable regulations, codes, and standards. New or revised regulations, codes, and standards are reviewed for incorporation into the QA Program and supporting manuals and procedures as necessary.

Personnel performing activities covered by the QA program shall perform work in accordance with approved procedures, and must demonstrate suitable proficiency in their assigned tasks. Formal training programs are established for quality assurance policies, requirements, procedures, and methods. Ongoing training is provided to ensure continuing proficiency as procedural requirements change. New employees are required to attend a QA indoctrination class on authority, organization, policies, manuals, and procedures.

Additional formal training is conducted in specific topics such as NRC regulations and guidance, procedures, auditing, and applicable codes and standards. Supplemental training is performed as required. On-the-job training is performed by the employee's supervisor in QA area-specific procedures and requirements. Training records are maintained for each person performing quality-related job functions.

The LES President assesses the scope, status, adequacy and regulatory compliance of the QA Program through regular meetings and correspondence with the Plant Manager and the LES

QA organization. Additionally, LES QA, through the QA Director, periodically informs the LES President and Plant Manager of quality concerns that need management resolution.

LES participates in the planning and scheduling for system turnover as construction is completed. Prior to system turnover, written procedures are developed for control of the transfer of systems, structures, components and associated documentation. The procedures include checklists, marked drawings, documentation lists, system status, and receipt control.

Major work activities contracted by LES shall be identified and controlled. Principal contractors shall be required to comply with the applicable portions of 10 CFR 50, Appendix B (CFR, 2003b), as determined by LES. The performance of contracted activities shall be formally evaluated by LES commensurate with the importance of the activities to safety.

Facility components and processes are assigned a QA level based on their safety significance. Each component will receive a classification of QA Level 1, QA Level 2, or QA Level 3 that applies throughout the life of the facility and is based on the following definitions:

QA Level 1 Requirements

The QA Level 1 Program shall conform to the criteria established in 10 CFR 50, Appendix B (CFR, 2003b). These criteria shall be met by commitments to follow the guidelines of ASME NQA-1-1994 (ASME, 1994), including supplements as revised by the ASME NQA-1a-1995 Addenda (ASME, 1995) as specified in the QA Program Description. The QA Level 1 QA program shall be applied to those structures, systems, components, and administrative controls that have been determined to be IROFS, items that affect the functions of the IROFS, and items required to satisfy regulatory requirements for which QA Level 1 requirements are applied.

QA Level 2 Requirements

The QA Level 2 program is an owner-defined QA program that uses the ASME NQA-1-1994 standard (ASME, 1994), including supplements as revised by the ASME NQA-1a-1995 Addenda (ASME, 1995) as guidance. General QA Level 2 requirements are described in Section 20, "Quality Assurance Program for QA Level 2 Activities". For contractors, the QA Level 2 program shall be described in documents that must be approved by LES. The QA Level 2 program shall be applied to Owner designated structures, systems, components, and activities. An International Organization for Standardization (ISO) 9000 series QA program may be acceptable for QA Level 2 applications provided it complies with LES Quality Assurance Program Description requirements. The QA program manual must be reviewed and accepted by the LES QA Director.

QA Level 3 Requirements

The QA Level 3 program is defined as standard commercial practice. A documented QA Level 3 program is not required. QA Level 3 governs all activities not designated as QA Level 1 or QA Level 2.

Appendix A, "LES Quality Assurance Program Description" of this chapter provides additional details and commitments to other QA elements that will be implemented to support the Management Measures described in this chapter.

11.9 REFERENCES

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APPENDIX A

Louisiana Energy Services

Quality Assurance Program Description

Design, Construction, Operations and Decommissioning Phases

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INTRODUCTION

Louisiana Energy Services (LES) maintains full responsibility for ensuring that the enrichment facility is designed, constructed, operated, and decommissioned in conformance with applicable regulatory requirements, specified design requirements, applicable industry standards and good engineering practices in a manner to protect the health and safety of the employees and the public. To this end, the LES Quality Assurance Program conforms to the criteria established in Title 10 of the Code of Federal Regulations 10 CFR 50, Appendix B, Quality Assurance Criteria For Nuclear Power Plants and Fuel Reprocessing Plants. The criteria in 10 CFR 50, Appendix B, are met by LES's commitment to follow the guidelines of the American Society of Mechanical Engineers (ASME) Quality Assurance (QA) standard NQA-1-1994, Quality Assurance Program Requirements for Nuclear Facilities, including supplements as revised by the ASME NQA-1a-1995 Addenda.

The LES QA Program described herein covers design, construction (including pre-operational testing), operation (including testing), maintenance and modification, and decommissioning of the facility. This Quality Assurance Program Description (QAPD) describes the requirements to be applied to those structures, systems and components, and activities that have been designated Quality Assurance (QA) Level 1.

QA Level 1 is applied exclusively to items relied on for safety (IROFS), any items which are determined to affect the function of the IROFS, and, in general, to items required to satisfy regulatory requirements. The development of the IROFS list is a product of the Integrated Safety Analysis (ISA) process. The Integrated Safety Analysis provides the methodology utilized to establish the IROFS list. IROFS are comprised of specific structures, systems and components (SSC) and administrative controls. All sections of this QAPD are applied to IROFS, any SSC and administrative controls which are determined to affect the functions of the IROFS and items required to satisfy regulatory requirements for which QA Level 1 requirements are applied. Application of the QAPD requirements is part of the configuration management system and will be performed in accordance with documented procedures. The LES QA organization reviews and concurs with the selection of the IROFS and the application of QA requirements to the IROFS, any items which are determined to affect the functions of the IROFS and items required to satisfy regulatory requirements for which QA Level 1 requirements are applied.

The QA Level 2 program description is provided in Section 20, Quality Assurance Program for QA Level 2 Activities of this QAPD. These requirements are implemented by LES and LES contractors through the use of approved QA programs and procedures. The Owner defined QA Level 2 SSCs and their associated activities i.e., those SSCs that are not IROFS, provide support of normal operations of the facility, and do not affect the functions of the IROFS (e.g., occupational exposure, radioactive waste management) and SSCs that minimize public, worker, and environmental risks (e.g., physical interaction protection, certain radiation monitors and criticality alarms) are evaluated against the requirements in Section 20, of this QAPD. This evaluation identifies which QA controls are needed to ensure these SSC meet their intended functions and do not affect the functions of the IROFS. This evaluation may also include nuclear industry precedent in the application of augmented QA requirements.

Three QA Levels have been established and apply throughout the life of the facility from licensing and siting through design, construction, operation, and decommissioning. The three levels are defined as follows.

QA LEVEL 1 REQUIREMENTS

The QA Level 1 Program shall conform to the criteria established in 10 CFR 50, Appendix B. These criteria shall be met by commitments to follow the guidelines of ASME NQA-1-1994, including supplements as revised by the ASME NQA-1a-1995 Addenda. The QA Level 1 QA program shall be applied to those structures, systems, components, and administrative controls that have been determined to be IROFS, items that affect the functions of the IROFS, and, in general, to items required to satisfy regulatory requirements.

QA LEVEL 2 REQUIREMENTS

The QA Level 2 program is an owner-defined QA program that uses the ASME NQA-1 standard as guidance. General QA Level 2 requirements are described in Section 20, Quality Assurance Program for QA Level 2 Activities. For contractors, the QA Level 2 program shall be described in documents that must be approved by LES. The QA Level 2 program shall be applied to Owner designated structures, systems, components, and activities. An International Organization for Standardization (ISO) 9000 series QA program may be acceptable for QA Level 2 applications provided it complies with applicable LES QAPD requirements and the QAPD is reviewed and accepted by the LES QA Director.

QA LEVEL 3 REQUIREMENTS

The QA Level 3 program is defined as standard commercial practice. A documented QA Level 3 program is not required. QA Level 3 governs all activities not designated as QA Level 1 or QA Level 2.

As described in Section 19, Provisions for Change, subsequent changes to the LES QA Program shall be incorporated in this QAPD. Any changes that reduce the commitments in the approved QAPD will be submitted to the Nuclear Regulatory Commission (NRC) for review and approval prior to implementation.

SECTION 1 ORGANIZATION

The elements of the LES QA Program described in this section and associated QA procedures implement the requirements of Criterion 1, Organization, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 1 and Supplement 1S-1 of NQA-1-1994.

LES employees and contractor employees representing LES have full responsibility to ensure that the facility is designed, constructed, operated, and decommissioned in a manner to protect the health and safety of the public. This responsibility begins with initial design and continues throughout the life of the facility. The LES QA Program is designed to ensure that the necessary quality requirements for structures, systems, components and work activities are met. This objective is attained by ensuring that the organizational structure and the responsibility assignments are such that (a) quality is achieved and maintained by those who have been assigned responsibility for performing work and, (b) quality achievement is verified by persons or organizations not directly responsible for performing the work.

CORPORATE ORGANIZATION AND FUNCTIONS

LES is the owner and operator of the enrichment facility. LES is a registered limited partnership formed to provide uranium enrichment services for commercial nuclear power plants. LES is responsible for the design, construction, operation and decommissioning of the enrichment facility in accordance with its QA Program. The President of LES reports to the LES Management Committee. The committee is composed of representatives from the general partners of LES.

The LES President establishes the basic policies of the QA Program. These policies are described in this QA Program, are transmitted to all levels of management, and are implemented through approved procedures. The LES QA Director has overall responsibility for development, management and implementation of the LES QA Program during all phases of the enrichment facility. As part of this responsibility, the QA Director is responsible for ensuring that contractor QA Programs meet all applicable requirements of the LES QA Program. LES management is continually involved in activities affecting quality and QA requirements.

Reporting to the President are the Engineering and Contracts Manager, Corporate Communications Manager, Chief Financial Officer (CFO), Quality Assurance Director, Chief Operating Officer (COO) and the Health, Safety and Environment Manager. Figure A1, LES Corporate, Design and Construction Organization, shows the levels of authority and lines of communications for activities affecting quality.

DESIGN AND CONSTRUCTION ORGANIZATION AND FUNCTIONS

The LES Engineering and Contracts Manager or the LES President acting in the capacity of the Engineering and Contracts Manager, has contracted Urenco, the owner of the enrichment technology and operator of enrichment facilities in Europe, to prepare the reference design for the facility. An architect/engineering (A/E) firm has been contracted and is under the responsibility of the Engineering and Contracts Manager or President to further specify structures and systems of the facility, and ensure the reference design meets all applicable U.S. codes and standards. A contractor specializing in site evaluations has been contracted and is under the responsibility of the Engineering and Contracts Manager or President to perform the site selection evaluation. A nuclear consulting company has been contracted and is under the responsibility of the Engineering and Contracts Manager or President to conduct the site characterization, perform the Integrated Safety Analysis and to support development of the license application including the Environmental Report.

During the design and construction phases, preparation of design and construction documents and construction itself are contracted to qualified contractors. The Engineering and Contracts Manager is responsible for managing the design, construction and construction inspection activities, startup, including pre-operational testing and procurement activities during these phases. Contractor QA Programs will be reviewed by the LES QA organization and must be approved by the LES QA Director before work can start as described in Section 4, Procurement Document Control, and Section 7, Control of Purchased Material, Equipment and Services. Urenco will design, manufacture and deliver to the site the centrifuges necessary for the facility under a QA Program approved by the LES QA Director or under the LES QA Program. In addition, Urenco is supplying the technical assistance and consultation for the facility in accordance with the applicable requirements of the LES QA Program. As shown in Figure A1, the Engineering and Contracts Manager is responsible for managing the work and contracts with the Technology Supplier (i.e.,Urenco) and a select group of Project Managers. These Project Managers will be responsible for the areas of Procurement, Construction, Engineering, Project Engineering, Project Controls and Start up.

QA Procedures will be developed by the Engineering and Contracts organization to implement this QAPD in the Engineering and Contracts area.

OPERATING ORGANIZATION AND FUNCTIONS

The operating organization is shown in Figure A2, LES National Enrichment Facility Operating Organization. The Plant Manager reports to the COO and is responsible for the overall operation and administration of the enrichment facility. The Plant Manager is also responsible for ensuring the facility complies with all applicable regulatory requirements including the requirements of this QAPD. In the discharge of these responsibilities, the Plant Manager directs the activities of the following groups.

- Health, Safety and Environment
- Operations
- Uranium Management
- Technical Services
- Human Resources
- Quality Assurance

Procedures will be developed by the respective operations organizations to implement the requirements of this QAPD. Specific details of organizational responsibilities and job descriptions are provided in the National Enrichment Facility (NEF) Safety Analysis Report.

QA ORGANIZATION AND FUNCTIONS

The LES QA organization during the design and construction phases will be headed by the LES QA Director. The LES QA Director reports directly to the LES President and is vested with the authority, access to work areas, and organizational independence to ensure that the requirements of this QAPD are properly implemented.

The LES QA Director is responsible for managing the LES QA Program that includes the following activities:

- QA Technical Support
  - o Maintain the LES QAPD
  - o Maintain QA procedures
  - o QA technical reviews of procurement documents
  - o Review and concurrence of changes to the identified IROFS, items that could affect the functions of IROFS, and items required to satisfy regulatory requirements for which QA Level 1 requirements are applied
  - o Administer the Corrective Action and Nonconformance Processes
  - o Maintain the LES Approved Suppliers List (ASL)
  - o Administer the Auditor and Lead Auditor Certification Process
  - o QA reviews of project documents
  - o Approval of contractor QA Programs
  - o Oversight of contractor QA Programs Implementation
  - o Oversight of the quality of design and construction, including but not limited to the ISA process and the resultant selection of IROFS
  - o Oversight of document and records control
- QA Verification
  - o Audits, surveillances and assessments
  - o Contractor/supplier evaluations
  - o Contractor nonconformances
  - o Equipment/Vendor Shop Inspections
  - o Witness vendor acceptance testing

During the transition from construction to operations, when startup testing and plant operations may be concurrent as the facility is completed in phases, a plant QA Manager will be added to the LES QA Organization. During this transition period as well as during operations, the plant QA Manager will report to the Plant Manager. However, the plant QA Manager has the authority and responsibility to contact the LES President, through the QA Director, with any QA concerns during startup and plant operations. After construction has been completed on the facility the corporate functions reporting the LES QA Director, i.e., QA Technical Support and QA Verification; will transition to the plant QA Manager. During the operations and decommissioning phases, the LES QA Director will advise the LES President on quality-related matters and continue to have governance and oversight responsibilities with respect to the QA organization headed by the plant QA Manager. The following additional QA Manager responsibilities are included for start up testing and operations:

- QA Technical Support
  - o Quality Engineering support of startup organization
  - o Oversight of startup activities

  - QA selected reviews and oversight of programs developed for operations, including but not limited to the ISA process, the identification of IROFS and items that affect the performance of IROFS and any changes thereto, the controls for assuring IROFS performance and verifying and maintaining the facility design basis.
  - QA selected reviews and oversight of operations including maintenance and testing and modification procedures
  - Review and concurrence of changes to the identified IROFS, items that could affect the functions of IROFS, and items required to satisfy regulatory requirements for which QA Level 1 requirements are applied
  - QA Oversight of operations procedure implementation
  - Quality Control (QC) Inspection certification process
- QC Inspections
  - Receipt Inspections of QA Level 1 items
  - Applicable discipline inspections of modifications to QA Level 1 components

Accordingly, during the transition from construction to operations, the operations phase, and the decommissioning phase, the management of the QA organization and the QA staff have the responsibility to make quality assurance decisions and have sufficient authority, access to work areas, and organizational freedom to:

- Identify quality problems
- Initiate and recommend solutions to quality problems through designated channels
- Verify implementation of solutions
- Assure that further processing, delivery, installation, or use of items is controlled until proper disposition of nonconformances, deficiencies or unsatisfactory conditions has occurred
- Have direct access to highest levels of management
- Be sufficiently independent from cost and schedule considerations and have stop-work authority.

ORGANIZATIONAL INTERFACES

The organizational interfaces between LES, contractors, and project applicable regulatory agencies are identified in the appropriate plans, contracts and implementing procedures. These documents contain the appropriate protocols, applicable roles, responsibilities and approval authorities for the specific topics for which they apply. LES design interfaces shall be identified and procedurally controlled. Design efforts shall be coordinated among interfacing organizations as detailed in LES procedures. Interface controls shall include the assignment of responsibility and the establishment of implementing documents among interfacing design organizations for the review, approval, release, distribution and revision of documents involving design interfaces. LES design information transmitted across interfaces shall be documented and procedurally controlled. LES transmittals of design information and/or documents shall reflect the status of the transmitted information and documents. Incomplete designs that require further evaluation, review or approval shall be identified. When it is necessary to initially transmit the design information orally or by other informal means, design information shall be promptly confirmed through a controlled implementing document.

DELEGATION OF WORK

The delegation of work between LES and contractors is identified in applicable plans, contracts and implementing procedures. In all cases of delegation, LES retains the overall responsibility for all work performed under the direction of LES. All LES QA Level 1 work activities shall meet the requirements of this QAPD. Responsible managers have the authority to delegate tasks to another qualified individual within their organization provided the designated individual possesses the required qualifications and these qualifications are documented. All delegations shall be in writing. The responsible manager retains the ultimate responsibility and accountability for implementing the applicable requirements.

RESOLUTION OF DISPUTES

Disputes involving differences of opinion on quality matters or issues are brought to the attention of line management, and if not resolved by the individual's manager, are elevated progressively to the QA Director. If satisfactory resolution cannot be obtained at that level, the matter is then elevated to the LES President for final resolution.

WORKER RESPONSIBILITIES

Each employee has an obligation to identify concerns using the corrective action process with respect to work within their scope of responsibility whenever the health and safety of our workers, the public, or the environment is involved or when continued work will produce results that are not in compliance with the LES QA Program. This process is controlled by an LES procedure, which applies across the entire project/facility. The authorities and responsibilities for stopping work, the criteria and documentation required to process the stop work and the actions required before work may resume are detailed in an LES procedure. This process ensures that safety related activities are controlled until the deficiency, or unsatisfactory condition, has been resolved. Worker responsibilities are further discussed in Section 16, Corrective Action.

SECTION 2 QUALITY ASSURANCE PROGRAM

The elements of the LES QA Program described in this section and associated QA procedures implement the requirements of Criterion 2, Quality Assurance Program, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 2 and Supplements 2S-1, 2S-2, 2S-3 and 2S-4 of NQA-1-1994 Part I as revised by NQA-1a-1995 Addenda of NQA-1-1994.

PROGRAM BASIS

The LES Quality Assurance Program complies with 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, and applies to all levels of the organization, including contractors, who perform QA Level 1 activities. Part I and selected sections of Part II of ASME NQA-1-1994, Quality Assurance Requirements for Nuclear Facility Applications, as revised by NQA-1a-1995 Addenda are used in conjunction with 10 CFR 50, Appendix B and provide additional detailed quality assurance guidelines which are committed to in this QAPD. The LES QAPD describes LES's overall compliance with 10 CFR 50, Appendix B and commitments to ASME NQA-1. This document states LES policies, assigns responsibilities and specifies requirements governing implementation of the QA Program to the design, construction, operation and decommissioning of the LES enrichment facility. All 18 criteria of 10 CFR 50, Appendix B have been addressed to identify the scope of QA Program applied to the LES enrichment facility. QA requirements will also apply to contractors as delineated in procurement documents controlled under Section 4, Procurement Document Control, of this QAPD. The necessary management measures to control the quality of subcontracted activities for the LES design, procurement, and installation and testing of QA Level 1 components and activities have been established in this QAPD. The QAPD will be reviewed for needed revisions as described in Section 19, Provisions For Change.

Specific processes and controls, which implement the provisions of 10 CFR 50, Appendix B and the commitment to ASME NQA-1-1994, as specified in this QAPD are delineated in procedures. Development, review, approval and training on procedures shall be performed prior to performance of the activities controlled by the procedures.

The QA Program provides for the planning and accomplishment of activities affecting quality under suitably controlled conditions. Controlled conditions include the use of appropriate equipment, suitable environmental conditions for accomplishing the activity, and assurance that prerequisites for the given activity have been satisfied. The LES QA Program provides for special controls, processes, test equipment, tools and skills to attain the required quality and verification of quality. QA requirements contained in this QAPD are also invoked on LES contractors for their contracted scope of work.

When work cannot be accomplished as specified in implementing QA procedures, or accomplishment of such work would result in an adverse condition, work is stopped until proper corrective action is taken. If procedures cannot be used as written, then work is stopped until the procedures are changed. Requirements for stop work are further discussed in Section 16, Corrective Action.

Flowdown of QA Requirements to Contractors and Suppliers

QA requirements for QA Level 1 activities are imposed on LES contractors and suppliers through the respective procurement documents for the particular scope of work being contracted. Determination of the specific QA requirements, supplier evaluations, and proposal/bid evaluations are in accordance with the requirements of Section 4, Procurement Document Control, and Section 7, Control of Purchased Material, Equipment and Services, of this document. Applicable QA Program elements required for the particular scope of work are identified in procurement documents. Potential contractors/suppliers are required to submit their QA Programs to the LES QA organization for review in accordance with the request for proposal/procurement specification. The LES QA organization performs an audit at the contractor's/supplier's facility of their QA program and its implementation verifying that the contractor's/supplier's QA program meets the requirements established in the request for proposal/procurement specification. If the audit is acceptable then the contractor/supplier is added to the LES ASL and a contract between LES and the contractor/supplier may be issued. For procured items, LES may also require that the LES QA organization perform source inspections or witness tests at the supplier's facility prior to shipment if the equipment/component warrants inspection due to its safety significance and/or complexity. Such requirements are also identified in the procurement documents and/or contract.

Construction contractors for LES QA Program controlled construction activities are required to be placed on the ASL prior to contract award. Construction contractors are required to perform the QA activities required by their QA program including audits of their own activities as well as any required quality control (QC) inspections. The LES QA organization will provide oversight of these contractors in the form of audits and surveillances verifying that each contractor is properly implementing its QA program as approved by LES QA. Contractually contractors will be required to promptly correct LES identified deficiencies and noncomformances.

IDENTIFICATION AND APPLICATION OF QA CONTROLS

QA Level 1 is applied exclusively to IROFS, any items which are determined to affect the function of the IROFS, and, in general, to items required to satisfy regulatory requirements. Since the development of the IROFS list is a product of the ISA process, the applicable QA Level 1 requirements are also applied to this process. The Integrated Safety Analysis provides the methodology utilized to establish the IROFS list. IROFS are comprised of specific structures, systems and components (SSC) and administrative controls. All applicable sections of this QAPD are applied to IROFS, any SSC and administrative controls which are determined to affect the functions of the IROFS and items required to satisfy regulatory requirements for which QA Level 1 requirements are applied. Application of the QAPD requirements is part of the configuration management program used to verify and maintain the facility design basis and will be performed in accordance with documented procedures. Accordingly, as described in Section 1, Organization, the QA organization is responsible for selected reviews and oversight of these processes and programs. In particular, the LES QA organization reviews and concurs with the selection of the IROFS and the application of QA requirements to the IROFS, any items which are determined to affect the functions of the IROFS and items required to satisfy regulatory reqirements for which QA Level 1 requirements are applied.

The QA Level 2 program description is provided in Section 20, Quality Assurance Program for QA Level 2 Activities of this QAPD. These requirements are implemented by LES and LES contractors through the use of approved QA programs and procedures. The Owner defined QA Level 2 SSCs and their associated activities i.e., those SSCs that are not IROFS, provide support of normal operations of the facility, and do not affect the functions of the IROFS (e.g., occupational exposure, radioactive waste management) and SSCs that minimize public, worker, and environmental risks (e.g., physical interaction protection, certain radiation monitors and criticality alarms) are evaluated against the requirements in Section 20, of this QAPD. This evaluation identifies which QA controls are needed to ensure these SSCs meet their intended functions and do not affect the functions of the IROFS. This evaluation may also include nuclear industry precedent in the application of augmented QA requirements.

Three QA Levels have been established and apply throughout the life of the facility from licensing and siting through design, construction, testing, startup, operation, maintenance, modification, and decommissioning. The three levels are defined as follows.

QA LEVEL 1 REQUIREMENTS

The QA Level 1 Program shall conform to the criteria established in 10 CFR 50, Appendix B. These criteria shall be met by commitments to follow the guidelines of ASME NQA-1-1994, including supplements as revised by the ASME NQA-1a-1995 Addenda. The QA Level 1 QA program shall be applied to those structures, systems, components, and administrative controls that have been determined to be IROFS, items that affect the functions of the IROFS, and, in general, to items required to satisfy regulatory requirements.

QA LEVEL 2 REQUIREMENTS

The QA Level 2 program is an owner-defined QA program that uses the ASME NQA-1 standard as guidance. General QA Level 2 requirements are described in Section 20, Quality Assurance Program for QA Level 2 Activities. For contractors, the QA Level 2 program shall be described in documents that must be approved by LES. The QA Level 2 program shall be applied to Owner designated structures, systems, components, and activities. An International Organization for Standardization (ISO) 9000 series QA program may be acceptable for QA Level 2 applications provided it complies with LES QAPD requirements and the QAPD is reviewed and accepted by the LES QA Director.

QA LEVEL 3 REQUIREMENTS

The QA Level 3 program is defined as standard commercial practice. A documented QA Level 3 program is not required. QA Level 3 governs all activities not designated as QA Level 1 or QA Level 2.

QUALITY ASSURANCE TRAINING

LES employees who perform QA Level 1 activities receive LES QA Indoctrination Training. This training includes general criteria, including introduction to applicable codes, standards, QA Procedures, QA Program elements and job responsibilities and authorities. LES personnel assigned to perform QA Level 1 activities are also required to complete training in the specific LES QA procedures needed to perform their job roles and responsibilities as assigned by their supervisor. Detailed QA training is provided on the LES QA Program and job specific QA procedures prior to an employee beginning QA Level 1 work. Supervision is responsible for ensuring that personnel performing work under their supervision are appropriately trained. LES will also include a version of QA Indoctrination Training as part of the general employee training given to all full-time employees.

The Human Resources Manager is responsible for coordinating QA training activities for LES. Human Resources serves as a centralized training support service for supervision in coordinating training and maintaining QA training records. This responsibility is carried out as support for line management. LES supervisory personnel are responsible for determining the type and extent of the training to be provided to an individual, and ensuring that the training is properly documented for personnel performing QA Level 1 activities. Retraining, when applicable, shall occur in order to maintain proficiency or when changes to work methods, technology, or job responsibilities occur. Such retraining is also documented.

MANAGEMENT ASSESSMENTS

The LES President is responsibility for ensuring that management assessments are conducted annually to determine if the LES QA Program is effective. Recommendations are provided to the LES President for action. Functional Managers and the QA Director conduct assessments annually of QA activities under their control. The managers report the results to the LES President for review. The results of these assessments are reviewed by senior management to determine the adequacy of implementation of the LES QA Program and to direct any needed changes for program improvements.

QUALIFICATION/CERTIFICATION OF INSPECTION AND TEST PERSONNEL

Inspection and test personnel performing QA Level 1 activities shall be certified in accordance with NQA-1-1994 Part I Supplement 2S-1, *Supplementary Requirements for the Qualification of Inspection and Test Personnel.*

QUALIFICATION/CERTIFICATION OF NONDESTRUCTIVE EXAMINATION (NDE) PERSONNEL

Nondestructive Examination (NDE) personnel performing QA Level 1 activities shall be certified in accordance with NQA-1a-1995 Part 1 Supplement 2S-2, *Supplementary Requirements for the Qualification of Nondestructive Examination Personnel and American Society of Nondestructive Testing (ASNT) Recommended Practice No. SNT-TC-1A, Personnel Qualification and Certification in Nondestructive Testing*, December 1988 Edition. Qualification/certification records meeting the requirements of Supplement 2S-2 shall be established and maintained as QA records.

QUALITY ASSURANCE AUDIT PERSONNEL

Audit personnel performing QA Level 1 activities shall be certified in accordance with NQA-1a-1995 Part 1 Supplement 2S-3 *Supplemental Requirements for the Qualification of Quality Assurance Program Audit Personnel.*

QUALITY ASSURANCE PROGRAM STATUS REPORTING TO MANAGEMENT

Management is regularly informed by the LES QA organization of adverse trends and lessons learned as a result of reviews conducted on audit reports, surveillance reports, corrective action reports, management assessments, etc. Corrective action is initiated as necessary.

SECTION 3 DESIGN CONTROL

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 3, Design Control, of 10 CFR 50, Appendix B, and the commitment to Basic Requirements 3 and Supplement 3S-1 of NQA-1-1994 Part I as revised by NQA-1a-1995 Addenda of NQA-1-1994. The LES QA Program also implements the commitment to Part II of NQA-1-1994 Subpart Part 2.7, *Quality Assurance Requirements of Computer Software for Nuclear Facility Applications,* as revised by NQA-1a-1995 Addenda of NQA-1-1994. These commitments also apply to computer software that is used to produce or manipulate data that is used directly in the design, analysis and operation of structures, systems and components relied on for safety. Part I, Supplement 11S-2, *Supplementary Requirements for Computer Program Testing*, requirements for computer software qualification and use are also implemented by the LES QA Program.

Measures are established in procedures to assure that applicable requirements are correctly translated into design documents. Design inputs are specified on a timely basis to support LES milestones. Controls are established for the selection and suitability of application of materials, parts, equipment and processes that are essential to the functions of structures, systems and components. Design interfaces to ensure completeness and efficiency of design are established in applicable procedures. Procedures detail the controls for design input, design process, design verification, design changes and approval. These procedures include appropriate quantitative and/or qualitative acceptance criteria for determining that activities have been satisfactorily accomplished. LES design documents are prepared, reviewed, and approved by qualified individuals. Design is verified by one or more of the following verification methods: design reviews, alternate calculations or qualification tests. Design changes are governed by control measures commensurate with those applied to the original design. The design process and design verification practices and procedures shall be reviewed and modified, as necessary, when a significant design change is required because of an incorrect design. These and any other design deficiencies discovered during the design process on subsequent design related activities that affect the design of SSC shall be entered into the Corrective Action Program (CAP) according to Section 16, Corrective Action. If these deficiencies cause constructed or partially constructed items (systems, structures or components) to be deficient, the affected items shall be controlled in accordance with Section 15, Nonconforming Items. Configuration management is maintained in accordance with the applicable procedure and the applicable procedures controlling changes to the various types of design documents.

DESIGN INPUT CONTROL

Applicable design inputs (such as design basis, conceptual design reports, performance requirements, regulatory requirements, codes and standards) shall be controlled by the LES Engineering and Contracts Manager according to the following requirements:

- Design inputs shall be identified and documented, and their selection reviewed and approved.
- Design inputs shall be specified and approved in a manner to support the schedule. Design inputs shall provide the necessary details to permit design to be carried out in a manner that provides a consistent basis for making design decisions, accomplishing design verification

and evaluating design changes.

- Changes from approved design inputs and reasons for the changes shall be identified, approved, documented and controlled.
- Design inputs based on assumptions that require re-verification shall be identified and controlled by the appropriate procedures.

DESIGN PROCESS

The LES design process shall be controlled by the Engineering and Contracts Manager according to the following requirements:

- LES design work shall be prescribed and documented on a timely basis and to the level of detail necessary to permit the design process to be carried out in a correct manner and to permit verification that the design meets requirements.
- Design documents shall be adequate to support design, construction and operation.
- Appropriate quality standards shall be identified and documented, and their selection reviewed and approved.
- Changes from specified standards, including the reasons for the change, shall be identified, approved, documented and controlled.
- Design methods, materials, parts, equipment and processes that are essential to the function of the structure, system, or component shall be selected and reviewed for and suitability of application.
- Applicable information derived from experience as set forth in reports or other documentation, shall be made available to cognizant design personnel.
- Final design documents (i.e., approved design output documents and approved changes thereto) shall be sufficiently detailed as to purpose, method, assumptions, design input, references and units such that a person technically qualified in the subject/engineering discipline can understand the documents and verify their adequacy without recourse to the originator of the design document.
- Procedural controls for identifying sub-assemblies or components on final design documents that are part of the item being designed shall be established. When a commercial grade item is modified and/or tested to new requirements that are different from the supplier's published product description, the component part shall be traceable to documentation noting that it is different from the originally approved commercial grade item.
- LES design drawings, specifications or other design output documents shall contain appropriate inspection, examination and testing acceptance criteria.

<u>DESIGN ANALYSIS</u>

LES design analyses shall be planned, controlled and documented. Design analysis documents shall be legible, in a form suitable for reproduction, filing and retrieval, and under configuration management control. LES design calculations shall be identifiable by subject (including structure, system or component to which the calculation applies), originator, reviewer and date, or by other designators in order that approved calculations are retrievable.

Computer software used to perform design analyses shall be developed and/or qualified, and used according to the provisions of ASME NQA-1-1994, Part II, Subpart 2.7 as revised by NQA-

1a-1995 Addenda and Supplement 11S-2. Computer software developed and/or qualified under the LES or its contractor QA programs may also be used to perform design analyses for LES, provided that the LES QA organization confirms these contractor QA programs meet the provisions NQA-1-1994, Part I, Supplement 11S-2 and NQA-1-1994 Part II, Subpart 2.7 as revised by NQA-1a-1995 addenda.

Computer programs may be utilized for design analysis without individual verification of the program for each application provided:

- The computer program has been verified to show that it produces correct solutions for the encoded mathematical model within defined limits for each parameter employed; and
- The encoded mathematical model has been shown to produce a valid solution to the physical problem associated with the particular application.

Computer programs shall be controlled to assure that changes are documented and approved by authorized personnel. Where changes to previously verified computer programs are made, verification shall be required for the change, including evaluation of the effects of these changes on the above.

LES design analyses documentation shall include:

- Definition of the objective of the analyses,
- Definition of design inputs and their sources,
- Results of literature searches or other applicable background data,
- Identification of assumptions and designation of those that must be verified as the design proceeds,
- Identification of any computer calculation, including computer type, computer program (e.g., name), revision identification, inputs, outputs, evidence of reference to computer program verification and the bases (or reference thereto) supporting application of the computer program to the specific physical problem,
- Review and approval.

DESIGN VERIFICATION

The following design control requirements shall be applied to verify the adequacy of LES design:

- LES design verification is required for design documents, and shall be performed using one or a combination of the design review, alternate calculations and/or qualification testing methods.
- The particular design verification method used shall be documented.
- Results of design verification shall be documented and shall include the identification of the verifier(s).
- Competent individuals or groups, other than those, who performed the original design (but may be from the same organization), shall perform design verification. If necessary, this verification may be performed by the originator's supervisor provided that the engineering supervisor did not specify a singular design approach or rule out certain design considerations and did not establish the design inputs used in the design; or the supervisor is the only individual in the organization competent to perform the verification.

LES design verification shall be performed in a timely manner at appropriate times during the design process. Verification shall be performed before release for procurement, manufacture or construction, or release to another organization for use in other design work. In some cases (such as when insufficient data exists) it may be necessary to release unverified designs to other engineering organizations or disciplines to support schedule requirements. Unverified portions of the design shall be clearly identified and procedurally controlled. In all cases, design verification shall be completed before relying on the item or computer program to perform its function. The extent of design verification required shall be a function of the importance to safety, complexity of design, degree of standardization, state of the art and similarity with previously proven designs.

LES use of previously standardized designs shall be controlled according to the following requirements:

- The applicability of standardized or previously proven designs shall be verified with respect to meeting pertinent design inputs for each application.
- Known problems affecting standard or previously proven designs and their effects on other features shall be considered.
- The "Americanization" of previously proven European designs shall be documented in accordance with the applicable QA procedure.
- The original design and associated verification measures shall be adequately documented and referenced in the files for subsequent application of the design.
- Changes in previously verified designs shall require re-verification. Such verifications shall include the evaluation of the effects of those changes on the overall previously verified design and on any design analyses upon which the design is based.

DESIGN VERIFICATION METHODS

Acceptable verification methods include, but are not limited to, any one of the following or a combination of the following:

- Design Reviews
- Alternate Calculations
- Qualification Testing

<u>DESIGN REVIEWS</u>

Design reviews are critical reviews to provide assurance that the final design is correct and satisfactory. The following items shall be addressed, as applicable during the review:

- Were the design inputs correctly selected and incorporated into the design?
- Are assumptions necessary to perform the design activity adequately described, reasonable and, where necessary, re-verified?
- Was an appropriate design method used?
- Is the design output reasonable compared to the applicable design inputs?
- Are the necessary design input and verification requirements for interfacing organizations specified in the design documents or in supporting procedures and instructions?

ALTERNATE CALCULATIONS

The appropriateness of assumptions, input data, and the computer program or other calculation methods used, shall be evaluated and the results shall be checked through the use of alternate calculation methods to verify the correctness of the original calculations or analyses.

QUALIFICATION TESTS

If design adequacy is to be verified by qualification testing, the tests shall be identified, procedurally controlled and documented according to the following:

- The test configuration shall be defined and documented.
- Testing shall demonstrate the adequacy of performance under conditions that simulate the most adverse design conditions. Operating modes and environmental conditions in which the item must perform satisfactorily shall be considered in determining the most adverse design conditions.
- If the tests verify only specific design features, then the other features of the design shall be verified by other means.
- Test results shall be documented and evaluated to ensure that test requirements have been met.
- If qualification testing indicates that a modification to an item is necessary to obtain acceptable performance, then the modification shall be documented and the item modified and re-tested or otherwise verified to ensure satisfactory performance.
- Scaling laws shall be established, verified and documented when tests are being performed on models or mockups.
- The results of model test work shall be subject to error analysis, where applicable, before using the results in final design work.

DESIGN CHANGE CONTROL

Design changes during the initial design phase and the operational phase shall be controlled according to the following requirements:

- Changes to final designs, field changes, modifications to the operating facility and nonconforming items dispositioned as "use-as-is" or "repair," as described in Section 15, Nonconforming Items, and shall have documented justification for use and are subject to the same design control measures and reviews as those applied to the original design.
- Design control measures for changes shall include provisions to ensure that the design analyses for the item are still valid.
- Changes shall be reviewed and approved by the affected groups or organizations that reviewed and approved the original design documents, with the following clarifications:
    - If the organization that originally was responsible for approving a particular design document is no longer responsible, then a new responsible organization shall be designated.
    - The designated organization shall have demonstrated competence in the specific design area of interest and have an adequate understanding of the requirements and intent of the original design.

- The interface between the design organization responsible for finalizing a design change and other organizations either involved in the review of the change, such as the QA and configuration management organizations, and those affected by the change, such as the operations and maintenance organizations, described in the next subsection, Design Interface Control, shall be maintained.
- The design process and design verification practices and procedures shall be reviewed and modified, as necessary, when a significant design change is required because of an incorrect design. These design deficiencies shall be documented according to Section 16.0, Corrective Actions. If these deficiencies cause constructed or partially constructed items (systems, structures or components) to be deficient, the affected items shall be controlled in accordance with Section 15, Nonconforming Items.
- When a design change is approved other than revision to the affected design documents, field changes shall be incorporated into affected design documents when such incorporation is appropriate.

DESIGN INTERFACE CONTROL

LES design interfaces shall be identified and procedurally controlled. Design efforts shall be coordinated among interfacing organizations as detailed in LES procedures. Interface controls shall include the assignment of responsibility and the establishment of procedures among interfacing design organizations for the review, approval, release, distribution and revision of documents involving design interfaces. LES design information transmitted across interfaces shall be documented and procedurally controlled. LES transmittals of design information and/or documents shall reflect the status of the transmitted information and documents. Incomplete designs that require further evaluation, review or approval shall be identified. When it is necessary to initially transmit the design information orally or by other informal means, design information shall be promptly confirmed through a controlled document.

During the operational phase, the Plant Manager is responsible for ensuring the facility complies with all applicable regulatory requirements including the requirements of this QA Program. In the discharge of these responsibilities, the Plant Manager directs the activities of the Technical Services, which includes Engineering and Maintenance, and Operations. Procedures for controlling the interfaces and configuration management ensure that changes and modifications are properly managed and disseminated to those responsible personnel or organizations whose duties may be affected by the design change or modification and do not adversely impact the safe operation of the plant.

COMPUTER SOFTWARE CONTROLS

If LES uses software to produce or manipulate data that is used directly in the design, analysis and operation of structures, systems, and components relied on for safety, the provisions provided in Part II ASME NQA-1-1994 Subpart Part 2.7, *Quality Assurance Requirements of Computer Software for Nuclear Facility Applications,* as revised by NQA-1a-1995 Addenda of NQA-1-1994 and ASME NQA-1-1994, Part I, Supplement 11S-2, *Supplementary Requirements for Computer Program Testing* shall apply. Procedures will be developed to implement of these provisions as applicable.

DOCUMENTATION AND RECORDS

Design documentation which provide evidence that the design and design verification were performed in accordance with this QAPD shall be collected and maintained in accordance with the requirements of Section 17 Quality Assurance Records. The documentation shall include not only final design documents such as drawings, specifications and revision thereto but also documentation which identifies the important steps, including sources of design inputs that support the final design.

SECTION 4 PROCUREMENT DOCUMENT CONTROL

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 4, Procurement Document Control, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 4 and Supplement 4S-1 of NQA-1-1994.

LES procurements shall be issued only to those suppliers that have been evaluated and qualified as acceptable for the particular scope of material, equipment and services to be procured. The material, equipment and services shall be procured from approved suppliers by procurement documents, approved by the LES President and QA Director or their qualified designees. Applicable design bases and other requirements necessary to assure adequate quality shall be included or referenced in documents for procurement of items and services. Procurement documents shall require suppliers to have a quality assurance program consistent with the applicable requirements of 10 CFR 50 Appendix B and this QAPD. The requirements of 10 CFR 21, Reporting of Defects and Noncompliance, are invoked during design, construction, testing and operations of QA Level 1 procurement or dedication of items and services including the dedication of items or services used to satisfy the requirements of 10 CFR 50, Appendix B or 10 CFR 70, Domestic Licensing of Special Nuclear Material. LES will also apply the requirements of 10 CFR 21 where appropriate, regardless of QA level.

Procurement Document Content

LES procurement documents issued for QA Level 1 items or services shall include the following provisions, as applicable to the procured material, equipment or service:

- Statement of the scope of work to be performed by the supplier.
- Technical requirements including:
    - Design bases, identified or referenced in the procurement documents.
    - Specific documents (such as drawings, codes, standards, regulations, procedures or instructions) describing the technical requirements of the material, equipment or services to be furnished, shall be specified along with their revision level or change status.
    - Tests, inspections or acceptance requirements that LES will use to monitor and evaluate the performance of the supplier shall be specified.
- Quality Assurance Program requirements including:
    - A requirement for the supplier to have a documented quality assurance program that implements applicable requirements of 10 CFR 50, Appendix B and this QAPD in place before the initiation of work. The extent of the quality assurance program shall depend on the scope, nature or complexity of the material, equipment or service to be procured. The supplier shall also incorporate the appropriate requirements into any subtier supplier issued procurement documents.
    - A requirement invoking NRC reporting requirements of 10 CFR 21 for QA Level 1 procurements.
- Right of access to supplier, including subtier, facilities and records for inspection or audit by LES, or other designee authorized by LES.
- Provisions for establishing witness/inspection hold points beyond which work cannot

proceed by the supplier without LES QA Director authorization. The LES Engineering and Contracts Manager may also establish hold points indicating work that cannot proceed without authorization by the Engineering and Contracts Manager.

- Documentation required to be submitted to LES for information, review or acceptance shall be identified along with a document submittal schedule. Record retention times, disposition requirements and record maintenance responsibilities shall be identified for documentation that will become quality assurance records.
- Requirements for the supplier to report to LES in writing adverse quality conditions resulting in work stoppages and nonconformances. LES approval of partial and full work releases and disposition of nonconformances is required.
- Identification of any spare and replacement parts or assemblies and the appropriate delineation of technical and quality assurance data required for ordering these parts or assemblies. Commercial Grade procurements shall also be identified in procurement documents.

Procurement Document Review and Approval

Procurement document reviews shall be performed and documented before issuing the procurement documents to the supplier. A review of the procurement documents and any changes thereto shall be made to verify that documents include all applicable requirements specified under Section 4, Procurement Document Content, above and contain appropriate provisions to ensure that material, equipment or services will meet the governing requirements. Reviews shall be performed and documented to provide objective evidence of satisfactory accomplishment of such review prior to contract award. Changes made as a result of the bid evaluation or precontract negotiations shall be incorporated into the procurement documents. The review of such changes and their effects shall be completed prior to contract award. This review shall include the following considerations: 1) appropriate requirements specified in Procurement Document Content above, 2) a determination of any additional or modified design criteria, and 3) an analysis of exceptions or changes requested by the supplier and a determination on the impacts such changes may have on the intent of the procurement documents or quality of the item or service to be provided shall be performed by the LES organization initiating the procurement. Personnel who have access to pertinent information and have an adequate understanding of the requirements and scope of the procurement shall perform reviews of the procurement documents. Reviewers shall include representatives from the Engineering and Contracts and QA organizations. The QA review shall assure compliance to quality assurance requirements.

Procurement Document Change

Changes to the scope of work, technical requirements, quality assurance program requirements, right of access, documentation requirements, work stoppage and nonconformance, hold points and lists of spare and replacement parts delineated in procurement documents, shall be subject to the same degree of control as used in the preparation of the original procurement document.

SECTION 5 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 5, Instructions, Procedures, and Drawings, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 5 of NQA-1-1994 Part I.

Activities affecting quality shall be prescribed by and conducted in accordance with approved procedures and other implementing documents (drawings, specifications, etc.) appropriate to the circumstances. Generally, four types of procedures are used by LES to ensure that activities are carried out in compliance with the requirements of this QAPD and in a safe manner. These include administrative, operating, maintenance and emergency procedures. Administrative procedures would include areas such as engineering procurement, etc. Administrative procedures are the higher level procedures that prescribe the implementation of the requirements provided in this QAPD. Operating and maintenance procedures are utilized to implement the QA program during the start up, operation, and testing of the facility. During the design and construction phases, procedures are reviewed and approved by the affected organizations with review and oversight by the QA organization. Those procedures that delineate the responsibilities and functions of the QA organization, the QA procedures, are approved by the LES QA Director to ensure compliance with QAPD. During operations, the LES QA Manager and Plant Manager have responsibility to review and approve the procedures that cover activities under their organizational purview that relate to the QAPD and the safe operation of the plant. Procedures approved by the Plant Manager will be subject to selected review and oversight by the QA organization.

TYPES OF DOCUMENTS

The type of document to be used to perform work shall be appropriate to the nature and circumstances of the work being performed. Documents include procedures, drawings and specifications. Work controlling procedures may also utilize approved checklists, travelers or other means to assure process requirements are met including prerequisite requirements prior to starting work. Procedures provide a consistent method for process performance and documentation of completion as well as ensure specified safety and environmental conditions are maintained.

CONTENT OF DOCUMENTS

Documents shall include or reference the following information as appropriate to the work to be performed:

- Responsibilities of the organizations affected by the document,
- Quality, technical and regulatory requirements,
- A sequential description of the work to be performed including controls for altering the sequence of required inspections, tests and other operations,
- Quantitative or qualitative acceptance criteria sufficient for determining that prescribed activities have been satisfactorily accomplished,
- Prerequisites, limits, precautions, process parameters and environmental conditions,
- Quality verification points and hold points,

- Methods for demonstrating that the work was performed as required,
- Identification of the lifetime or nonpermanent quality assurance records generated by the implementing document, and
- Identification of associated QA Levels as appropriate.

REVIEW, APPROVAL, AND CONTROL OF DOCUMENTS

Procedures and implementing documents shall be controlled according to the requirements of Section 6, Document Control of this document. Procedures and implementing documents shall be reviewed and approved as described in this section and in Section 6.

SECTION 6 DOCUMENT CONTROL

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 6, Document Control, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 6 and Supplement 6S-1 of NQA-1-1994.

Procedures are established which control the preparation, issuance and changes of documents that specify quality requirements or prescribe activities affecting quality. Measures are established to ensure that documents, including revisions are adequately reviewed, approved, and released for use by authorized personnel. Controlled documents are transmitted to the appropriate locations where the prescribed activity is being performed. Superceded documents are destroyed or retained only when they have been properly marked.

TYPES OF DOCUMENTS

QA procedures, other administrative procedures and implementing documents and documents specifying quality requirements or prescribing activities affecting quality shall be controlled in accordance with this section. LES documents controlled under the LES QA Program will be specified by procedures and include, but are not limited to, procedures, design requirements document, design basis documents, engineering specifications, instructions, drawings, calculations, procurement documents, and documents that need to be controlled due to being input to other LES design documents or used for construction and operations affecting quality.

PREPARING AND REVIEWING DOCUMENTS

The document control system shall ensure that the identification of documents to be controlled and their specified distribution are preceduralized. The system shall further ensure that the responsibility for preparing, reviewing, approving and issuing documents shall be assigned by procedure to the appropriate LES functional area manager. Implementing documents and documents specifying quality requirements or prescribing activities affecting quality, shall be reviewed in accordance with applicable procedures for adequacy, correctness and completeness and by the QA organization as specified by procedure, prior to approval and issuance. The organizational position(s) responsible for approving the document(s) for release shall be identified in the applicable procedures.

CONTROLLING THE DISTRIBUTION AND USE OF DOCUMENTS

Documents needing to be placed under the document control system are transmitted to the Document Control organization with the distribution list for document holders. The Document Control organization shall enter the document into the Document Control electronic database and master list of controlled documents, assign document control numbers, complete transmittal forms and distribute the documents and transmittal form to the document holders. Document holders shall acknowledge receipt on the transmittal and send the acknowledgement to the Document Control organization. The up-to-date master listing of controlled documents will be made continuously available to document holders to verify that they have the current revisions. The document control process will be audited in accordance with the requirements of Section 18, QA Audits, to verify implementation effectiveness.

CHANGES TO DOCUMENTS

Changes to documents other than minor changes shall be reviewed for adequacy, correctness and completeness, prior to approval and issuance. Major changes shall be reviewed and approved by the same organization that performed the original review and approval unless other organizations are specifically designated. The reviewing organization shall have access to the applicable background data or information upon which to base their approval. A temporary procedure change that does not change the intent of the procedure may be made at the work location by responsible management. The applicable procedure shall control the process, documentation and approval of the temporary changes.

MINOR CHANGES

Minor changes such as inconsequential editorial corrections may be made to documents without being subject to the review and approval of the requirements specified above. The applicable procedure shall define the organizational positions authorized and criteria acceptable for making minor changes.

SECTION 7 CONTROL OF PURCHASED MATERIAL, EQUIPMENT AND SERVICES

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 7, Control of Purchased Material, Equipment and Services, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 7 and Supplement 7S-1 of NQA-1-1994 Part I as revised by NQA-1a-1995 Addenda of NQA-1-1994.

LES procurement of material, equipment and services is controlled to assure conformance with specified requirements. These controls include requirements for pre-award evaluations of suppliers' QA programs, annual evaluations, periodic audits/source inspections and surveillance. Suppliers with a LES approved QA program are placed on the LES ASL prior to award of contract. Source inspections and surveillances, evaluation of objective evidence of quality furnished by the supplier, maintaining the ASL, as well as, examination of received items and services are the responsibility of LES QA organization and are performed, as necessary, upon delivery or completion to ensure requirements specified in procurement documents are met. Supplier evaluations, annual evaluations, audits, surveillances, source inspections and receipt inspections shall be documented.

PROCUREMENT PLANNING

LES procurements shall be planned and documented to ensure a systematic approach to the procurement process exists and supports the schedule. Procurement planning shall:

- Identify procurement methods and organizational responsibilities, including what is to be accomplished, who is to accomplish it, how it is to be accomplished, and when it is to be accomplished.
- Identify and document the sequence of actions and milestones needed to effectively complete the procurement.
- Provide for the integration of the following activities:
  - Procurement document preparation, review and change control according to the requirements of Section 4, Procurement Document Control
  - Selection of procurement sources, proposal/bid evaluation and award
  - LES evaluation of supplier performance
  - LES verifications including any hold and witness point notifications
  - Control of nonconformances
  - Corrective action
  - Acceptance of the material, equipment or service
  - Identification of quality assurance records to be provided to LES.
- Be accomplished as early as possible, and no later than at the start of those procurement activities that are required to be controlled to assure interface compatibility and a uniform approach to the procurement process.
- Be performed relative to the level of importance, complexity and quantity of the item or service being procured and the supplier's quality performance.

- Include the involvement of the LES QA organization to ensure that the QA requirements have been properly identified.

SOURCE EVALUATION AND SELECTION

Supplier selection shall be based on an evaluation, performed before the contract and/or purchase order is awarded, of the supplier's capability to provide items or services in accordance with procurement document (technical and quality) requirements. The functional area needing the procurement shall request that the LES QA organization evaluate the potential supplier for placement on the LES ASL. Responsibilities and measures for evaluating and selecting procurement sources are detailed in the applicable QA procedure and include one or more of the following methods for evaluating potential suppliers:

- Evaluation of the supplier's history for providing an identical or similar product that performs satisfactorily in actual use. The supplier's history shall reflect current capability.
- Evaluation of supplier's current quality assurance records supported by any documented *qualitative and quantitative information which can be objectively evaluated.*
- Evaluation of the supplier's technical and quality capability based on an evaluation of supplier facilities, personnel and quality assurance program implementation.

The results of procurement source evaluation and selection shall be documented in accordance with the applicable QA procedure.

PROPOSAL/BID EVALUATION

For proposals and bids, technically qualified personnel from the QA and Engineering and Contracts or other affected/involved organizations shall perform an evaluation to determine if the proposal/bid meets procurement document requirements. As a minimum, this evaluation shall review the following subjects consistent with the importance, complexity and quantity of items or services being procured:

- Technical considerations
- QA program requirements
- Supplier personnel qualifications
- Supplier production capability and past performance
- Alternatives and exceptions

Before the contract is awarded, the LES QA Director or Engineering and Contracts Manager, or other affected/involved organization manager shall resolve, or obtain commitments to resolve, unacceptable quality conditions identified during the proposal/bid evaluation. Supplier quality assurance programs shall be evaluated by the QA organization before contract placement, and any deficiencies that would affect quality shall be corrected before starting work subject to these requirements. Supplier QA programs shall be accepted by the LES QA Director before the supplier starts work.

SUPPLIER PERFORMANCE EVALUATION

The LES Engineering and Contracts Manager in coordination with the QA Director shall establish measures to routinely interface with the supplier and to verify supplier performance. The measures shall include:

- Establishing an understanding between LES and the supplier of the requirements and specifications identified in procurement documents.
- Requiring the supplier to identify planning techniques and processes to be used in fulfilling procurement document requirements.
- Reviewing supplier documents that are prepared or processed during work performed to fulfill procurement requirements.
- Identifying and processing necessary change information.
- Establishing the method to be used to document information exchanges between LES and supplier.
- Establishing the extent of source surveillance and inspection.

The extent of LES verifications shall be a function of the relative importance, complexity/quantity of items or services being procured and the supplier's quality performance. Verification activities shall be accomplished by qualified personnel assigned to check, inspect, audit, or witness the activities of the suppliers. LES verifications shall be conducted as early as practical and shall not relieve the supplier of the responsibility for the verification of quality achievement. Verifications shall include supplier audits, surveillances or source inspections (or combinations) used as a method of evaluating the supplier's performance, and evaluation of purchaser's documentation to aid in the determination of the effectiveness of the supplier's quality assurance program. Records, including source surveillances and inspections, audits, receiving inspections, nonconformances, dispositions, waivers, and corrective actions shall be maintained in accordance with the requirements of Section 17, Quality Assurance Records.

CONTROL OF SUPPLIER GENERATED DOCUMENTS

Supplier generated documents shall be controlled, processed and accepted by LES in accordance with the requirements established in the applicable QA procedures. Measures shall be implemented to ensure that the submittal of supplier generated documents is accomplished in accordance with the procurement document requirements. These measures shall also provide for the acquisition, processing and recorded evaluation of technical, inspection and test data compared against the acceptance criteria.

CONTROL OF CHANGES IN ITEMS OR SERVICES

LES shall establish contractual controls with suppliers to ensure that changes in procurement documents are controlled and documented in accordance with this QAPD.

ACCEPTANCE OF ITEMS OR SERVICES

Methods for accepting supplier furnished material, equipment or services shall include one or more of the following, as appropriate to the items or services being procured:

- Evaluating the supplier certificate of conformance,
- Performing one or a combination of source verification, receiving inspection or post-installation test,

- Technical verification of the data produced (services only),
- Surveillance or audit of the activities (services only),
- Review of objective evidence for conformance to procurement requirements (services only).

The supplier shall verify that furnished material, equipment or services comply with LES's procurement requirements before offering the material, equipment or services for acceptance and shall provide to LES objective evidence that material, equipment or services conform to procurement documents. Where required by code, regulations or contract provisions, documentary evidence that items conform to procurement documents shall be available at the site prior to installation or use.

CERTIFICATE OF CONFORMANCE

When a certificate of conformance is used to accept material, equipment or service:

- The certificate shall identify the purchased material, equipment or service to the specific procurement document.
- The certificate shall identify the specific procurement requirements met by the purchased material, equipment or service. The procurement requirements identified shall include any approved changes, waivers or deviations applicable to the material, equipment or service.
- The certificate shall identify any procurement requirements that have not been met together with an explanation and the means for resolving nonconformances.
- The certificate shall be signed and dated or otherwise authenticated by an individual who is responsible for the supplier's quality assurance function and whose responsibilities and position are described in the supplier's quality assurance program.
- The certification process, including the implementing documents to be followed in filling out a certificate and the administrative implementing documents for review and approval of the certificates, shall be described in the supplier's quality assurance program.
- Measures shall be identified to verify the validity of supplier certificates and the effectiveness of the certification process (such as by audit of the supplier or by an independent inspection or test of the item). Verifications shall be conducted by LES at intervals commensurate with the past quality performance of the supplier.

SOURCE VERIFICATION

LES may accept material, equipment or service by monitoring, witnessing or observing activities performed by the supplier. This method of acceptance is called source verification. Source verification shall be implemented consistent with the supplier's planned inspections, examinations or tests at predetermined points and performed at intervals consistent with the importance and complexity of the item. Documented evidence of acceptance of source verified material, equipment or services shall be furnished to the receiving destination of the item, to LES, and to the supplier. Personnel qualified in accordance with the applicable requirements for the material, equipment or service being procured shall perform source verification.

RECEIVING INSPECTION

When receiving inspection is used to accept an item:

- The inspection shall consider any source verifications/audits and the demonstrated quality performance of the supplier.

- The inspection shall be performed in accordance with established inspection procedures.
- The inspection shall verify, as applicable, proper configuration; identification; dimensional, physical and other characteristics; freedom from shipping damage; and cleanliness.
- The inspection shall be planned and executed according to the requirements of Section 10 Inspection.
- Receiving inspection shall be coordinated with a review for adequacy and completeness of any required supplier documentation submittals.

POST-INSTALLATION TESTING

When post-installation testing is used as a method of acceptance, the LES Engineering and Contracts Manager or the affected/involved LES organization manager and the supplier, when possible, shall mutually establish test requirements and acceptance documentation. The LES Engineering Contracts Manager is ultimately responsible for ensuring appropriate test requirements and acceptance documentation are established.

CONTROL OF SUPPLIER NONCONFORMANCES

The LES Engineering and Contracts organization and the supplier shall establish and document the process for disposition of items that do not meet procurement document requirements. The supplier shall evaluate nonconforming items according to the applicable requirements of Section 15, Nonconforming Items and submit a report of nonconformance to LES Engineering and Contracts organization including supplier recommended disposition (for example, use-as-is or repair) and technical justification. Reports of nonconformances to procurement document requirements, or documents approved by LES, shall be submitted to LES Engineering and Contracts organization for approval of the recommended disposition whenever one of the following conditions exists:

- Technical or material requirements are violated.
- A requirement in supplier documents, which have been approved by LES, is violated.
- The nonconformance cannot be corrected by continuation of the original manufacturing process or by re-work.
- The item does not conform to the original requirement even though the item can be restored to a condition such that the capability of the item to function is unimpaired.

LES Engineering and Contracts organization shall disposition the supplier's recommendation and verify implementation of the disposition. LES will maintain records of the supplier-submitted nonconformances.

COMMERCIAL GRADE ITEMS

Where the design utilizes commercial grade material and/or equipment, the following requirements are an acceptable alternate to other requirements of this Section:

- The commercial grade material/equipment is identified in an approved design output document. An alternate commercial grade material/equipment may be applied, provided there is verification that the alternate commercial grade material/equipment will perform the intended function and will meet design requirements applicable to both the replaced material/equipment and its application.

- Supplier evaluation and selection, where determined necessary by the LES based on complexity and importance to safety, shall be in accordance with *Source Evaluation and Selection* section of this document.
- Commercial grade items shall be identified in the purchase order by the manufacturer's published product description (e.g., catalog number).
- One or a combination of the following methods shall be utilized to provide reasonable assurance that the item meets the acceptance criteria for the characteristics identified to be verified for acceptance:
  - special test(s) or inspection (s) or both;
  - commercial grade survey of the supplier;
  - source verification;
  - acceptable supplier/item performance records.
- Prior to acceptance of a commercial grade item, LES QA organization shall determine that:
  - damage was not sustained during shipment;
  - the item received has satisfied the specified acceptance criteria;
  - inspection and/or testing is accomplished, as required, to assure conformance with critical characteristics; and
  - documentation, as applicable to the item, was received and is acceptable.

APPROVED SUPPLIER LIST

The LES Quality Assurance Director is responsible for the development and maintenance of the LES ASL. The ASL contains those suppliers with acceptable QA Programs that have been evaluated and accepted by the LES QA in accordance with approved procedures. The LES QA organization shall perform and document an evaluation of each supplier every 12 months. Satisfactory results will allow the supplier to remain on the ASL. Additionally, suppliers will be evaluated by means of an audit at least triennially, if initial approval was by audit or survey. Suppliers that have unacceptable evaluations or that have not had a procurement placed with them in three years will be removed from the ASL.

SECTION 8 IDENTIFICATION AND CONTROL MATERIALS, PARTS AND COMPONENTS

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 8, Identification and Control of Materials, Parts and Components, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 8 and Supplement 8S-1 of NQA-1-1994 Part I as revised by NQA-1a-1995 Addenda.

The controls necessary to ensure that only correct and accepted items are used or installed will be required by the appropriate QA procedure. Identification requirements for materials, parts and components are stated in design specifications, drawings, and procurement documents. Specific identification requirements are as follows.

- Identification markings, when used shall be applied using materials and methods which provide a clear and legible identification and do not detrimentally affect the function or service life of the item. Markings shall be transferred to each part of an item when subdivided and shall not be obliterated or hidden by surface treatments or coatings unless other means of identification are substituted.
- When required by specifications or codes and standards, identification of material or equipment with traceability to the corresponding mill test reports, certifications and other required documentation is maintained throughout fabrication, erection, installation, or use.
- Sufficient precautions shall be taken to preclude identifying materials in a manner that degrades the function or quality of the item being identified.

Control of material, parts and components is governed by approved procedures. Specific control requirements include the following.

- Identification of nonconforming or rejected materials, parts or components to ensure that they are not inadvertently used.
- Verification of correct identification of materials (including consumable materials or items with a limited shelf life), parts, and components shall be required to prevent the use of incorrect or defective items.
- Receipt inspection to ensure that materials, parts or components are properly identified and that supporting documentation is available as required by procurement specifications.
- Maintaining and replacement of markings and identification records due to damage during handling, aging or environmental exposure.

SECTION 9 CONTROL OF SPECIAL PROCESSES

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 9, Control of Special Processes, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 9 and Supplement 9S-1 of NQA-1994 Part I.

Processes affecting the quality of items or services shall be controlled by written procedures using drawings, checklists, travelers or other appropriate means. These means shall ensure that the process parameters are controlled and that specified environmental conditions are maintained. Special processes that control or verify quality, such as those used in welding, heat treating, and nondestructive examination, shall be performed by qualified personnel using qualified procedures in accordance with specified requirements.

SPECIAL PROCESSES

Special processes that control or verify quality shall be controlled according to the requirements of this section whether or not they are covered by existing codes and standards, or whether or not the quality requirements specified for an item exceed those of existing codes or standards.

PERSONNEL, IMPLEMENTING DOCUMENTS, AND EQUIPMENT QUALIFICATIONS

Implementing LES documents shall be used to ensure that process parameters are controlled and that the specified environmental conditions are maintained. Each special process shall be performed in accordance with appropriate implementing documents and these implementing documents shall include or reference:

- The responsibility of the organization performing the special process to adhere to the approved procedures and processes,
- Qualification requirements for personnel, implementing documents and equipment,
- Conditions necessary for accomplishment of the special process. These conditions shall include proper equipment, controlled parameters of the process and calibration requirements, and/or
- Requirements of applicable codes and standards, including acceptance criteria for the special process.

QUALIFICATION OF NONDESTRUCTIVE EXAMINATION PERSONNEL

Personnel who have been qualified and certified in accordance with Section 2.0, QA Program, of this QAPD shall perform nondestructive examinations required for the LES work activities.

DOCUMENTATION

Records shall be maintained as appropriate in accordance with Section 17, Quality Assurance Records, for currently qualified personnel, processes and equipment of each special process.

SECTION 10 INSPECTION

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 10, Inspection, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 10 and Supplement 10S-1 of NQA-1-1994 Part I.

Inspections required to verify conformance of an item or activity to specified requirements are planned and executed. Characteristics to be inspected and inspection methods to be employed are specified in procedures. Inspection results are documented. Persons other than those who performed or directly supervised the work being inspected shall perform inspection for acceptance. Inspection requirements and acceptance criteria shall include specified requirements contained in the applicable design documents or other pertinent technical documents approved by the responsible design organization. Inspection activities are documented and controlled by instructions, procedures, drawings, checklists, travelers or other appropriate means.

INSPECTION PLANNING

Inspection planning shall be performed, documented and include:

- Identification of each work operation where inspection is necessary to ensure quality and implementing documents that shall be used to perform the inspections;
- Identification of the characteristics to be inspected and the identification of when, during the work process, inspections are to be performed;
- Identification of inspection or process monitoring methods to be employed;
- The final inspection shall be planned to arrive at a conclusion regarding conformance of the item to specified requirements;
- Identification of the functional qualification level (category or class) of personnel performing inspections;
- Identification of acceptance criteria;
- Methods to record objective evidence of inspection results; and
- Selection and identification of the measuring and test equipment to be used to perform the inspection.

SELECTING INSPECTION PERSONNEL TO PERFORM INSPECTION

The individual who performs an inspection to verify conformance of an item to specified acceptance criteria shall be qualified to perform the assigned inspection tasks in accordance with the requirements of Section 2, QA Program. Data recorders, equipment operators or other inspection team members who are supervised by a qualified inspector shall not be required to be a qualified inspector. Verification of conformance shall be by a qualified person. Inspections shall be performed by personnel other than those who performed or directly supervised the work being inspected. Inspection personnel shall not report directly to the immediate supervisors who are responsible for performing the work being inspected.

INSPECTION HOLD POINTS

When mandatory hold points are used to control work that shall not proceed without the specific consent of the organization placing the hold point, the specific hold points shall be indicated in implementing documents. Consent to waive specified hold points shall be documented and approved before continuing work beyond the designated hold point.

STATISTICAL SAMPLING

When statistical sampling is used to verify the acceptability of a group of items, the statistical sampling method used shall be based on recognized standard practices and these practices shall be implemented through applicable approved procedures.

IN-PROCESS INSPECTIONS AND MONITORING

Items shall be inspected when necessary to verify quality. If inspection of processed items is impossible or disadvantageous, indirect control by monitoring of processing methods, equipment and personnel shall be provided. Inspection and process monitoring shall be conducted when control is inadequate with only one method. A combination of inspection and process monitoring methods, when used, shall be performed in a systematic manner to ensure that the specified requirements for control of the process and the quality of the item are met throughout the duration of the process. Controls shall be established and documented for the coordination and sequencing of inspections and monitoring at established inspection points during successive stages of the process or construction.

FINAL INSPECTION

Finished items shall be inspected for completeness, markings, calibration, adjustments, protection from damage or other characteristics as required in order to verify the quality and conformance of the item to specified requirements. Documentation not previously examined shall be examined for adequacy and completeness. The final inspection shall be planned to arrive at a conclusion regarding conformance of the item to specified requirements. Final inspections shall include a review of the results and resolution of any nonconformances identified by earlier inspections. Modifications, repairs or replacements of items performed subsequent to final inspection shall require re-inspection or retest, as appropriate, to verify acceptability.

ACCEPTING ITEMS

The acceptance of an item shall be documented and approved by qualified and authorized personnel. The inspection status of an item shall be identified according to Section 14, Inspection, Test and Operating Status.

INSERVICE INSPECTION

Inservice inspection or surveillance of structures, systems, or components shall be planned and implemented by or for the LES Operating organization. Procedures shall control the inspections to verify that the characteristics of the item remain with the specified limits. The inspection procedure shall include the following, as appropriate:

- Evaluations of performance capabilities of essential emergency and safety systems and equipment,
- Verification of calibration and integrity of instruments and instrument systems, and
- Verification of maintenance.

INSPECTION DOCUMENTATION

Inspection documentation shall identify:

- The item inspected, date of inspection, the name of the inspector who documented, evaluated and determined acceptability;
- Name of data recorder, as applicable and type of observation or method of inspection;
- The inspection criteria, sampling plan or reference documents (including revision levels) used to determine acceptance;
- Results or acceptability of characteristics inspected;
- Measuring and test equipment used during the inspection including the identification number and the most recent calibration date; and
- Reference to information on actions taken in connection with nonconformances, as applicable.

SECTION 11 TEST CONTROL

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 11, Test Control, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 11 and Supplement 11S-1 of NQA-1-1994 Part I. The commitment to the provisions in Supplement 11S-2, Supplementary Requirements for Computer Program Testing is addressed in Section 3, Design Control.

Tests required to verify conformance of an item or computer program to specified requirements and to demonstrate satisfactory performance for service are planned and executed. Characteristics to be tested and test methods to be employed are specified. Test results are documented and their conformance with acceptance criteria is evaluated. Tests required to collect data, such as for siting or design input, shall be planned, executed, documented and evaluated.

TEST REQUIREMENTS

Test requirements and acceptance criteria shall be provided or approved by the organization responsible for the design of the item to be tested unless otherwise designated. Required tests, including, as appropriate, prototype qualification tests, production tests, proof tests prior to installation, construction tests, pre-operational tests, and operational tests are controlled. Test requirements and acceptance criteria are based upon specified requirements contained in applicable design or other pertinent technical documents.

TEST PROCEDURES

Test procedures shall include:

- Test objectives and the identification of any implementing documents to be developed to control and perform tests as appropriate;
- Identification of items to be tested, test requirements and acceptance limits, including required levels of precision and accuracy;
- Identification of test methods to be employed and instructions for performing the test;
- Test prerequisites that address calibrated instrumentation, appropriate and adequate test equipment/instrumentation, trained personnel, condition of test equipment and the item to be tested, suitably controlled environmental conditions and provisions for data acquisition;
- Mandatory hold points and methods to record data and results;
- Provisions for ensuring that prerequisites for the given test have been met;
- Selection and identification of the measuring and test equipment to be used to perform the test to ensure that the equipment is of the proper type, range, accuracy, and tolerance to accomplish the intended function; and
- Identification of the functional qualification level of personnel performing tests.

PERFORMING TESTS

Tests shall be performed in accordance with procedures that address the following requirements as applicable:

- Provisions for determining when a test is required, describing how tests are performed, and ensuring that testing is conducted by trained and appropriately qualified personnel.
- Include or reference test objectives and provisions for ensuring that prerequisites for the given test have been met, adequate calibrated instrumentation is available and used, necessary monitoring is performed and suitable environmental conditions are maintained.
- Test requirements and acceptance criteria provided or approved by the organization responsible for the design of the item to be tested, unless otherwise designated.
- Test requirements and acceptance criteria based upon specified requirements contained in applicable design or other pertinent technical documents.
- Potential sources of uncertainty and error. Test parameters affected by potential sources of uncertainty and error shall be identified and controlled.

MONITORING AND OVERSIGHT OF SUPPLIER TEST

The LES Engineering and Contracts Manager in coordination with the QA Director shall establish measures to routinely interface with the supplier and to verify supplier performance. LES may accept material, equipment or service by monitoring, witnessing or observing activities performed by the supplier. This method of acceptance is called source verification. Source verification shall be implemented consistent with the supplier's planned inspections, examinations or tests at predetermined points and performed at intervals consistent with the importance and complexity of the item. Documented evidence of acceptance of source verified material, equipment or services shall be furnished to the receiving destination of the item, to LES, and to the supplier. Personnel qualified in accordance with the applicable requirements for the material, equipment or service being procured shall perform source verification.

USE OF OTHER TESTING DOCUMENTS

Other testing documents (e.g., American Society for Testing and Materials (ASTM)) specifications, supplier manuals or other related documents containing acceptance criteria may be used instead of preparing special test procedures. If used, the information shall be incorporated by reference in the approved test procedure. Implementing documents shall include adequate supplemental instructions as required to ensure the required quality of the testing work.

TEST RESULTS

Test results shall be documented and their conformance with acceptance criteria shall be evaluated by a qualified individual within the responsible organization to ensure that test requirements have been satisfied.

TEST DOCUMENTATION

Test documentation shall include:

- Item or work product tested, date of test, names of tester and data recorders, type of observation and method of testing;
- Identification of test criteria or reference documents used to determine acceptance;
- Results and acceptability of the test;
- Actions taken in connection with any nonconformances or deviations noted;
- Name of the person evaluating the test results; and
- Identification of the measuring and test equipment (M&TE) used during the test.

SECTION 12 CONTROL OF MEASURING AND TEST EQUIPMENT

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 12, Control of Measuring and Test Equipment, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 12 and Supplement 12S-1 of NQA-1-1994 Part I.

This section establishes LES control for tools, gages, instruments and other measuring and test equipment (M&TE) used for activities affecting quality, including design activities where applicable, construction, operation and decommissioning. M&TE is controlled and at specified periods calibrated and adjusted to maintain accuracy within necessary limits. Selection of M&TE shall be controlled to ensure that such items are of proper type, range, accuracy, and tolerance to accomplish the functions of determining conformance to specified requirements.

CALIBRATION

M&TE shall be calibrated, adjusted and maintained at prescribed intervals or, prior to use, against reference calibration standards having traceability to nationally recognized standards. If no nationally recognized standards or physical constants exist, the basis for calibration shall be documented. Calibration standards shall have a greater accuracy than the required accuracy of the M&TE being calibrated. If calibration standards with a greater accuracy than required of the M&TE being calibrated do not exist or are unavailable, calibration standards with accuracy equal to the required calibration accuracy may be used, provided they are shown to be adequate for the requirements. The basis for the calibration acceptance shall be documented and authorized by responsible management as defined in applicable procedures. The level of management authorized to perform this function shall be identified. The method and interval of calibration for each device shall be defined, based on the type of equipment, stability characteristics, required accuracy, intended use and other conditions affecting measurement control. For M&TE used in one- time-only applications, the calibration shall be performed both before and after use. A calibration shall be performed when the accuracy of calibrated M&TE is suspect. Calibrated M&TE shall be labeled, tagged, or otherwise suitably marked or documented to indicate due date or interval of the next calibration and uniquely identified to provide traceability to its calibration data.

DOCUMENTING THE USE OF M&TE

The use of M&TE shall be documented. As appropriate to equipment use and its calibration schedule, the documentation shall identify the processes monitored, data collected or items inspected or tested since the last calibration.

OUT OF CALIBRATION M&TE

M&TE shall be considered to be out-of-calibration and not be used until calibrated if any of the following conditions exist:

- The calibration due date or interval has passed without re-calibration.
- The device produces results known or suspected to be in error.

- Out-of-Calibration M&TE shall be controlled. The controls shall include the following requirements:
    - o Out-of-Calibration M&TE shall be tagged, segregated or otherwise controlled to prevent use until they have been recalibrated.

When M&TE is found out-of-calibration, the validity of results obtained using that equipment since its last valid calibration shall be evaluated to verify the acceptability of previously collected data, processes monitored, or items previously inspected or tested. The evaluation shall be documented.

If any M&TE is consistently found out-of-calibration during the re-calibration process, it shall be repaired or replaced.

LOST M&TE

When M&TE is lost, the validity of results obtained using that equipment since its last valid calibration shall be evaluated to determine acceptability of previously collected data, processes monitored or items previously inspected or tested. The evaluation shall be documented.

HANDLING AND STORAGE

M&TE shall be properly handled and stored to maintain accuracy.

COMMERCIAL DEVICES

Calibration and control shall not be required for rulers, tape measures, levels and other normal commercial equipment that provides adequate accuracy.

M&TE DOCUMENTATION

M&TE calibration documentation shall include the following information:

- Identification of the measuring or test equipment calibrated;
- Traceability to the calibration standard used for calibration;
- Calibration data;
- Identification of the individual performing the calibration;
- Identification of the date of calibration and the re-calibration due date or interval, as appropriate;
- Results of the calibration and statement of acceptability;
- Reference to any actions taken in connection with out-of-calibration or nonconforming M&TE including evaluation results, as appropriate; and
- Identification of the implementing document used in performing the calibration.

SECTION 13 HANDLING, STORAGE, AND SHIPPING

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 13, Handling, Storage and Shipping, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 13 and Supplement 13S-1 of NQA-1-1994 Part I.

Handling, storage, cleaning, packaging, shipping and preservation of items are controlled in accordance with requirements of this section to prevent damage or loss and to minimize deterioration.

CONTROLS

Handling, storage, cleaning, packaging, shipping and preservation of items shall be conducted in accordance with established work and inspection implementing procedures, shipping instructions or other specified documents. For critical, sensitive, perishable or high-value articles, specific instructions for handling, storage, cleaning, packaging, shipping and preservation shall be prepared and used.

SPECIAL EQUIPMENT, TOOLS AND ENVIRONMENTS

If required for particular items, special equipment (i.e., containers, shock absorbers and accelerometers) and special protective environments (i.e., inert gas and specific moisture/temperature levels) shall be specified and provided. If special equipment and environments are used, provisions shall be made for their verification. Special handling tools and equipment shall be used and controlled as necessary to ensure safe and adequate handling. Special handling tools and equipment shall be inspected and tested at specified time intervals and in accordance with procedures to verify that the tools and equipment are adequately maintained. Operators of special handling and lifting equipment shall be experienced or trained in the use the equipment.

MARKING AND LABELING

Measures shall be established for marking and labeling for the packaging, shipping, handling and storage of items as necessary to adequately identify, maintain and preserve the item. Markings and labels shall indicate the presence of special environments or the need for special controls if necessary.

SECTION 14 INSPECTION, TEST, AND OPERATING STATUS

The elements of the LES QA Program described in this section and associated procedures implement the requirements of Criterion 14, Inspection, Test and Operating Status, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 14 of NQA-1-1994 Part I.

This section establishes requirements for LES to identify the status of inspection and test activities. Status is indicated either on the items or in documents traceable to the items where it is necessary to assure that required inspections and tests are performed and to assure that items which have not passed the required inspections and tests are not inadvertently installed, used or operated. Status is maintained through indicators (i.e., physical location and tags, markings, shop travelers, stamps, inspection records or other suitable means). The authority for application and removal of tags, markings, labels and stamps are specified. Status indicators shall also provide for indicating the operating status of systems and components of the nuclear facility (i.e., tagging valves and switches) to prevent inadvertent operation.

Process control procedures, test and inspection procedures, nonconforming item control procedures, installation records, and checklists are used as applicable to control the installation of structures, system and components. These documents contain hold points, activity checklists, and in many cases, step-by-step signoffs which indicate the status of fabrication, installation, inspections, and test. This system is used to prevent inadvertent use of nonconforming items or bypassing of inspections and tests and prevent inadvertent operation.

During operation, in order to ensure that equipment status is clearly evident, and to prevent inadvertent operation, the LES QA Program requires structures, systems and components that are inoperable to be identified as such. This identification may be by means of tags, labels, stamps or other suitable methods. When tags, labels, or stamps are utilized for the identification of equipment status, the issuance and removal thereof is documented to ensure proper control of such identification measures. Also, procedures require that the operability of an item removed from operation for maintenance or testing be verified prior to returning the item to normal service.

Measures taken by QA personnel, during the performance of required inspection and quality control activities, to identify equipment status are controlled by the QA organization independent of measures taken to identify and control equipment status by LES.

Changing the sequence of inspections, tests, and other activities involving safety requires the same controls as the original review and approval.

SECTION 15 NONCONFORMING ITEMS

The elements of the LES QA Program described in this section and associated QA procedures implement the requirements of Criterion 15, Nonconforming Items, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 15 and Supplement 15S-1 of NQA-1-1994 Part 1.

This section provides the process for controlling items that do not conform to specified requirements. For the purposes of this QAPD, items referenced to in this section means materials, parts, or components. The control of nonconforming activities and services is described in Section 16, Corrective Action. These items are controlled to prevent inadvertent installation or use. The controls provide for identification, documentation, evaluation, segregation when practical, disposition of nonconforming items and for notification to affected organizations.

DOCUMENTING AND EVALUATING NONCONFORMING ITEMS

Nonconformance documentation shall clearly identify and describe the characteristics that do not conform to specified criteria. Nonconformance documentation shall be reviewed by the responsible affected organization and recommended dispositions of nonconforming items shall be proposed in accordance with procedures. The review shall include determining the need for additional corrective actions according to the requirements of Section 16, Corrective Action. In addition, organizations affected by the nonconformance shall be notified. Recommended dispositions shall be evaluated and approved in accordance with procedures. Personnel performing evaluations of recommended dispositions shall have demonstrated competence in the specific area they are evaluating, an adequate understanding of the requirements and access to pertinent background information. The responsibility and authority for reviewing, evaluating, approving the disposition and closing nonconformances shall be specified in procedures. The LES QA Organization is responsible for administering the Nonconformance Process. QA can initiate, recommend, or provide solutions via designated channels. QA will verify the implementation of the corrective actions and QA will assure that procedures are in place to control the installation and use of nonconformances until an acceptable solution has been provided. Further processing, delivery, installation or use of a nonconforming item shall be controlled pending the evaluation and approval of the disposition by authorized personnel.

IDENTIFYING NONCONFORMING ITEMS

Employees of LES and LES contractors have a procedural obligation to identify and document nonconformances. Nonconforming items shall be identified by marking, tagging or other methods that do not adversely affect their end use. The identification shall be legible and easily recognizable. If the identification of a nonconforming item is not practical, the container, package or segregated storage area, as appropriate, shall be identified.

SEGREGATING NONCONFORMING ITEMS

Nonconforming items shall be segregated, when practical, by placing them in a clearly identified and designated hold area until properly dispositioned. If segregation is impractical or impossible due to physical conditions, then other precautions shall be employed to preclude inadvertent use.

DISPOSITION OF NONCONFORMING ITEMS

The disposition, such as "use-as-is," "reject," "repair," or "rework," of nonconforming items shall be identified and documented. The technical justification for the acceptability of a nonconforming item that has been dispositioned "repair" or "use-as-is" shall be documented.

Items that do not meet original design requirements that are dispositioned "use-as-is" or "repair" shall be subject to design control measures commensurate with those applied to the original design. If changes to the specifying document are required to reflect the as-built condition, the disposition shall require action to change the specifying document to reflect the accepted nonconformance. Any document or record change required by the disposition of the nonconformance shall be identified in the nonconformance documentation; and, when each document or record is changed, the justification for the change shall identify the nonconformance documentation. The disposition of an item to be reworked, or repaired shall contain a requirement to reexamine (inspect, test, or nondestructive examination) the item to verify acceptability. Repaired or reworked items shall be reexamined in accordance with applicable procedures using the original process and acceptance criteria unless the nonconforming item disposition has established alternate acceptance criteria.

TRENDING

Nonconformance documentation shall be periodically analyzed by the LES QA organization to identify adverse quality trends in accordance with Section 16, Corrective Action.

SECTION 16 CORRECTIVE ACTION

The elements of the LES QA Program described in this section and associated QA procedures implement the requirements of Criterion 16, Corrective Action, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 16 of NQA-1-1994 Part 1.

Conditions adverse to quality including activities and services shall be identified promptly and corrected as soon as practical. For significant conditions adverse to quality, the cause of the condition shall be determined and corrective action taken to preclude recurrence. The identification, cause, and corrective action for significant conditions adverse to quality shall be documented and reported to appropriate levels of management. Follow-up action shall be taken to verify implementation of the corrective action. Significant conditions adverse to quality shall be tracked and evaluated so that adverse trends can be identified and appropriate corrective action can be taken.

Procedure(s) shall be issued to establish the CAP which includes the following processes, including closure:

- Prompt identification and correction of conditions adverse to quality;
- Evaluating significant conditions adverse to quality for reportability to the NRC (when required) under 10 CFR 21, Reporting of Defects and Noncompliance, or other applicable reporting requirements and reporting such conditions when warranted;
- Stopping work, if applicable;
- Determining root cause and corrective actions to preclude recurrence for significant conditions adverse to quality; and
- Follow-up actions to verify implementation of corrective actions taken for significant conditions adverse to quality.

IDENTIFYING AND CLASSIFYING CONDITIONS ADVERSE TO QUALITY

Conditions adverse to quality shall be classified in one of two categories in regard to their significance, and corrective actions shall be taken accordingly. The two categories of significance include:

- Conditions adverse to quality
- Significant conditions adverse to quality

Conditions adverse to quality are defined as failures, malfunctions, deficiencies, deviations, defective material and equipment and nonconformances. Conditions adverse to quality shall be documented and reported to the appropriate levels of management.

Responsible management shall investigate and fully identify the condition and document the results. Responsible management shall then utilize investigation results to determine and document corrective action (including remedial action and if appropriate, actions to prevent recurrence). Responsible management shall complete remedial action and document completion of actions in a timely manner.

Significant conditions adverse to quality are defined as:

- A deficiency that would seriously impact an item, activity or service from meeting or performing its intended function or output of assuring public health and safety;

- A deficiency in design that has been approved for fabrication or construction where the design deviates extensively from design criteria and bases;
- A deficiency in the fabrication or construction of, or significant damage to, structures, systems or components that require extensive evaluation, re-design or repair in order to establish the adequacy of the structure, system or component to perform its intended function of assuring public health and safety;
- A deviation from performance specifications that shall require extensive evaluation, re-design, or repair to establish the adequacy of the structure, system or component to perform its intended function;
- A significant error in a computer program used to support activities affecting quality after it has been released for use;
- A deficiency, repetitive in nature, related to an activity or item subject to the LES QA Program; and
- A condition that, if left uncorrected, has the potential to have a serious negative impact on activities or items subject to the LES QA Program controls.

If a supplier or subtier supplier discovers a defect or noncompliance which the supplier evaluates as a substantial safety hazard, then the supplier shall be required to report the item under 10 CFR 21, Reporting of Defects and Noncompliance, and notify the LES in writing. If the supplier or subtier supplier is unable to determine if the defect/non compliance is a substantial safety hazard then the supplier or subtier supplier is required to report the item to LES for determination of reportability.

Significant conditions adverse to quality shall be evaluated for a stop work condition to determine if stopping work is warranted. If a stop work condition is identified, management shall issue stop work in accordance with the applicable procedure. Upon resolution of the related significant condition adverse to quality, management shall take appropriate action to lift and close (in part or total) the stop work order.

FOLLOW-UP ACTION

The procedure(s) establishing the Corrective Action Program shall include a requirement for management to take follow-up action to verify implementation of corrective action taken to address significant conditions adverse to quality. The QA organization shall be responsible for conducting periodic assessments of these follow-up actions.

TRENDING

The procedure(s) establishing the CAP shall assign organizational responsibility for trending significant conditions adverse to quality and the criteria for determining trends. Reports of significant conditions adverse to quality shall be evaluated to identify adverse quality trends and help identify root causes. Trend evaluation shall be performed in a manner and at a frequency that provides for prompt identification of adverse quality trends. Identified adverse trends shall be handled in accordance with the CAP described here and reported to the appropriate management.

SECTION 17 QUALITY ASSURANCE RECORDS

The elements of the LES QA Program described in this section and associated QA procedures implement the requirements of Criterion 17, Quality Assurance Records, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 17 and Supplement 17S-1 of NQA-1-1994 Part I.

A QA record is any completed record that furnishes documentary evidence of the quality of items and/or activities affecting quality. Records may include specially processed records such as radiographs, photographs, negatives, microforms and magnetic/electronic media. LES completed QA records that furnish documentary evidence of quality shall be specified, prepared and maintained in accordance with applicable regulatory requirements and applicable procedures. QA Records shall be legible, identifiable, retrievable, and shall be protected against damage, deterioration and loss. Requirements and responsibilities for record transmittal, distribution, retention, maintenance and disposition shall be established and documented in procedures. Retention periods for the various types of records generated under the LES QA Program shall be specified as Lifetime or Nonpermanent according to the criteria provided in this Section. The term "records" used throughout this section is to be interpreted as "Quality Assurance Record," unless otherwise specified.

RECORD MANAGEMENT SYSTEM

LES shall establish a record management system and LES Records Center at the earliest practicable time consistent with the schedule for accomplishing work activities and in compliance with the requirements of this QAPD. The QA records management system shall be defined, implemented and enforced in accordance with written procedures, instructions or other documentation. Records shall be distributed, handled, and controlled in accordance with written procedures.

GENERATION, CLASSIFICATION AND RETENTION OF QA RECORDS

Applicable LES design specifications, procurement documents, test procedures, operational procedures or other documents and procedures shall specify the records to be generated, supplied or maintained. Documents that are designated to become records shall be legible, accurate and completed appropriate to the work accomplished. LES records shall be classified for retention purposes as lifetime records or nonpermanent records in accordance with the criteria provided below.

- Lifetime records are those that meet one or more of the following criteria:
  - Those which would be of significant value in demonstrating capability for safe operation;
  - Those which would be of significant value in maintaining, reworking, repairing, replacing or modifying an item;
  - Those which would be of significant value in determining the cause of an accident or malfunction of an item; and/or
  - Those which provide required baseline data for in-service inspections.

Lifetime records are required to be maintained for the life of the particular item while it is installed in the facility or stored for future use.

Nonpermanent records are those required to show evidence that an activity was performed in accordance with the applicable requirements of the LES QA Program but need not be retained for the life of the item because they do not meet the criteria for lifetime records. The retention period for nonpermanent records shall be documented in the applicable procedure.

Procedures shall identify those documents that will become QA records. The individual using the procedure is responsible for ensuring the QA records required by the procedure are submitted to the LES Records Center. Documents that may become records shall be maintained and processed in a prudent manner to avoid unnecessary delay and/or expense in retrieving the record when the record is needed to support other work.

Individuals creating records shall ensure the records are legible, accurate and complete, and shall protect them from damage, deterioration or loss during the time the records are in their possession.

Documents shall be considered valid records only if authenticated (i.e., stamped, initialed or signed and dated complete by authorized personnel). If the nature of the record precludes stamping or signing, then other means of authentication by authorized personnel is permitted. This may take the form of a statement by the responsible individual or organization. Handwritten signatures are not required if the document is clearly identified as a statement by the reporting individual or organization. QA records may be originals or copies. LES contractors shall submit to the LES Records Center those records being temporarily stored by them in accordance with contractual requirements. The timing of the submittal shall be as records become completed, or as items are released for shipment, or as prescribed by QA procedures and procurement documents. Records shall be controlled and submitted to the records management system in accordance with implementing procedures.

RECEIVING QA RECORDS

Each organization responsible for receiving records shall provide protection from damage or loss during the time that the records are in their possession. A receipt control system shall be established by the organization to include the following:

- A method for designating the required records;
- A method for identifying records received;
- Procedures for receipt and inspection of incoming records; and
- A method for submittal of completed records to the storage facility without unnecessary delay; and
- Capability to provide current and accurate status of records during the receipt process.

Records shall be indexed to ensure retrievability. Records and/or indexing systems shall provide sufficient information to permit identification between the record and the item or activity to which it applies. The indexing system shall include:

- The location of the records within the records management system;
- Identification of the item or related activity to which the records pertain; and
- The retention classification of the record.

STORING, SAFEKEEPING, AND PRESERVING QA RECORDS

Records shall be stored and preserved in the LES Records Center in accordance with a procedure that includes the following:

- Assignment of responsibility for enforcing the requirements of the procedure;
- A description of the storage facility;
- A description of the filing system to be used;
- A method for verifying that the records received are in agreement with the transmittal document;
- A method for verifying that the records are those designated and the records are legible and complete;
- A description of rules governing control of the records, including access, retrieval and removal;
- A method for maintaining control of and accountability for records removed from the storage facility;
- A method for filing supplemental information and disposition of superseded records;
- A method for precluding entry of unauthorized personnel into the storage area to guard against larceny and vandalism; and
- A method for providing for replacement, restoration or substitution of lost or damaged records.

Storage methods shall be approved by the organization responsible for storage to preclude deterioration of records in accordance with the following:

- Provisions shall be made in the storage arrangement to prevent damage from moisture, temperature and pressure.
- Approved filing methods shall require records to be firmly attached in binders, or placed in folders or envelopes, for storage in steel file cabinets or on shelving in containers appropriate for the record medium being stored.
- The storage arrangement shall provide adequate protection of special processed records (e.g., radiographs, photographs, negatives, microform and magnetic media) to prevent damage from humidity, temperature, excessive light, electromagnetic fields or stacking, consistent with the type of record being stored.

LES RECORDS CENTERS

Originating organizations shall store records in temporary storage while active and required for use; subsequently the records shall be transmitted for permanent storage in accordance with the requirements of this Section and associated procedures.

LES organizations shall provide for temporary storage of records during processing, review or use, until turnover to the LES Records Center for disposition, according to implementing procedures and the following requirements:

- Records shall be temporarily stored in a container or facility with a fire rating of one (1) hour. The temporary storage container or facility shall bear an Underwriters' Laboratories label (UL) (or equivalent) certifying one (1) hour fire protection, or be certified by a person competent in the technical field of fire protection.

- The maximum time limit for keeping records in temporary storage shall be specified by implementing procedures consistent with the nature or scope of work.

LES QA records permanent storage shall either invoke the alternate single storage facility provision of Section 4.4.2 and/or the dual facilities provision of Section 4.4.4 of Supplement 17S-1 of NQA-1-1994. With either provision used, the LES Records Center shall be constructed and maintained in a manner that minimizes the risk of damage or destruction from the following:

- Natural disasters (i.e., winds, floods or fires);
- Environmental conditions (i.e., high and low temperatures and humidity); and
- Infestation of insects, mold or rodents.

If the alternate single storage facility provision is used, then LES records shall be stored in the LES Records Center in two (2) hour fire rated Class B file containers meeting the requirements of National Fire Protection Association (NFPA) 232-1986 or NFPA 232AM-1986 or both.

If the dual storage facility provision is used for hard copies, then LES records shall be stored with one copy in the LES Records Center and the second copy stored in facility that is sufficiently remote from the Records Center to eliminate the chance of exposure to a simultaneous hazard. If the dual storage facilities provision is used via scanned documents into an electronic records management system, then a back-up tape shall be periodically made of the electronic records management system and its contents and the tape shall be stored in atemporary storage devise in a fire-proof safe. This process invokes the dual storage provision as one copy resides on the records management system computer and a second copy of the total records system resides in a remote location with temporary storage being used for records entered in the interim.

RETRIEVING AND DISPOSITIONING QA RECORDS

The records management system shall provide for retrieval of records in accordance with planned retrieval times based upon the designated record type. Access to records storage facilities shall be controlled. A list shall be maintained designating personnel who are permitted access to the records at the LES Records Center.

Records maintained by a supplier at its facility or other location shall be accessible to the purchaser or designated alternate. The supplier's records shall not be disposed of until contractual requirements are satisfied.

Records accumulated at various locations prior to transfer shall be made accessible to LES directly or through the procuring organization. The record-keeper shall inventory the submittals, acknowledge receipt and process these records in accordance with this QAPD. Various regulatory agencies have requirements concerning records that are within the scope of this Section. The most stringent requirements should be used in determining the final disposition. The supplier's nonpermanent records shall not be disposed of until the applicable conditions listed below are satisfied.

- Items are released for shipment, a Code Data Report is signed, or a Code Symbol stamp is affixed.
- Regulatory requirements are satisfied.

- Operational status permits.
- Warranty consideration is satisfied.
- Purchaser's requirements are satisfied.

RETENTION OF QA RECORDS

Lifetime records shall be retained and preserved for the operating life of the particular item while it is installed in the plant or stored for future use. Nonpermanent records shall not be disposed of until the following conditions are met:

- Regulatory requirements are satisfied;
- Facility status allows document disposal; and
- LES QAPD requirements are satisfied

CORRECTING INFORMATION IN QA RECORDS

Corrections shall include the identification of the person authorized to make the correction and the date the correction was made. Corrections to records shall be performed in accordance with implementing procedures, which provide for appropriate review or approval of the corrections, by the originating organization.

REPLACING LOST OR DAMAGED QA RECORDS

Replacement, restoration or substitution of lost or damaged records shall be performed in accordance with implementing procedures, which provide for appropriate review or approval by the originating organization and any additional information associated with the replacement.

SECTION 18 AUDITS

The elements of the LES QA Program described in this section and associated QA procedures implement the requirements of Criterion 18, Audits, of 10 CFR 50, Appendix B, and the commitment to Basic Requirement 18 and Supplement 18S-1 of NQA-1-1994 Part 1.

In accordance with the description of the QA organization during the various phases of design, construction, and operation provided in Section 1, Organization, the LES QA Director or QA Manager shall verify LES compliance with all aspects of the LES QA Program and determine QA Program effectiveness by ensuring that planned and scheduled audits are conducted. Elements that have been selected for audit shall be evaluated against specified requirements. An auditing function reports to the LES QA Director/QA Manager and has the organizational independence and authority to execute an effective audit process to meet all requirements of the QA Program. Objective evidence shall be examined to the depth necessary to determine if these elements are being implemented effectively. LES audits are performed in accordance with written procedures or checklists by appropriately trained and qualified personnel who do not have direct responsibility for performing the activities being audited. Audit results are documented and provided to the appropriate management for review and corrective action as applicable. Follow-up actions are taken where indicated.

AUDIT SCHEDULES

Internal or external audits shall be scheduled in a manner to provide coverage, consistency and coordination with ongoing work, and at a frequency commensurate with the status and importance of the work. Internal or external audits shall be scheduled to begin as early in the life of the work as practical and shall be scheduled to continue at intervals consistent with the schedule for accomplishing the work. As a minimum, internal audits of LES QA Level 1 activities shall be at least once per year or at least once during the life of the activity, whichever is shorter. Regularly scheduled internal audits shall be supplemented by additional audits of specific subjects when necessary to provide an adequate assessment of compliance or effectiveness. Internal audits to determine quality assurance program effectiveness shall be performed on selected work products. The audit schedule shall be developed annually and revised as necessary to ensure that coverage is maintained current. Frequency of audits should be based upon evaluation of all applicable and active elements of the LES QAPD applicable to LES workscope. These evaluations should include an assessment of the effectiveness of the applicable and active elements of the LES QAPD based upon previous audit results and corrective actions, nonconformance reports, identified trends, and significant organizational changes.

AUDIT PLANS

A documented audit plan shall be developed for each audit. This plan shall identify the audit scope, requirements for performing the audit, type of audit personnel needed, work to be audited, organizations to be notified, applicable documents, audit schedule, and implementing documents or checklists to be used.

AUDIT TEAMS

The LES QA Director or QA Manager shall select and assign auditors who are independent of any direct responsibility for performing the work being audited. Audit personnel shall have sufficient authority and organizational freedom to make the audit process meaningful and effective. The audit team shall include one or more auditors comprised of representatives from the LES QA organization and any applicable technical organizations. A lead auditor shall be appointed to supervise the team, organize and direct the audit, prepare and coordinate issuance of the audit report and evaluate responses. Technical specialists may be used to assist in assessing the adequacy of technical processes. Before commencing the audit, the lead auditor shall ensure the personnel assigned to the audit team are prepared and collectively have experience and/or training commensurate with the scope, complexity or special nature of the work to be audited. Lead auditors, auditors and technical specialists shall be trained and qualified according to the requirements of Section 2, Quality Assurance Program.

PERFORMING AUDITS

The LES QA Director or QA Manager shall provide written notification of a planned audit to the affected organizations at a reasonable time before the audit is to be performed. The notification should include all relevant information pertaining to the audit, such as schedule, scope and names of audit lead and team members, if known. In addition, the audit team leader shall ensure the following is performed.

- The audit team shall be adequately prepared before starting the audit.
- Audits shall be performed in accordance with written procedures or checklists.
- Elements that have been selected for the audit shall be evaluated against specified requirements.
- Objective evidence shall be examined to the depth necessary to determine if the selected elements are being implemented effectively.
- Audit results shall be documented by auditing personnel, and reported to/reviewed by management having responsibility for the area audited. Conditions requiring prompt corrective action shall be reported immediately to management of the audited organization.
- Identified audit findings shall be documented and the audited organization shall correct the findings according to the requirements of Section 16, Corrective Action. Minor audit findings can be corrected during the conduct of the audit.

REPORTING AUDIT RESULTS

The audit report shall be prepared and signed by the audit team leader and issued to the management of the audited organization in a timely manner after completion of the audit.

The audit report shall include the following information:

- A description of the audit scope.
- Identification of the auditors.
- Identification of persons contacted during the audit.
- A summary of audit results and the documents reviewed, persons interviewed and the specific results of the reviews and interviews (i.e., a summary of the checklist contents).

- Statement as to the effectiveness of the implementation of the QA Program elements audited.
- A description of each reported adverse audit finding in sufficient detail to enable corrective action to be taken by the audited organization.
- A requested date for response by the audited organization.

RESPONDING TO AUDITS

Management of the audited organization or activity shall:

- Investigate adverse audit findings in a timely manner;
- Determine and schedule corrective action, including measures to prevent recurrence;
- Prior to or by the requested response date, notify the LES QA Director in writing of the actions taken or scheduled, according to the requirements of Section 16 Corrective Action.

EVALUATING AUDIT RESPONSES

The LES QA Director or QA Manager is responsible for evaluating audit responses.

FOLLOW-UP ACTION

Follow-up action shall be taken by the LES QA Director to verify that:

- Corrective actions are completed as scheduled according to the requirements of Section 16 Corrective Action.

RECORDS

- Audit records include audit plans and audit reports.
- Written replies and the record of completion of any required corrective actions.

These documents are QA records and shall be submitted to the LES Records Center for retention according to the requirements of Section 17, Quality Assurance Records.

NON-LES AUDITOR QUALIFICATIONS

Non-LES certified auditors may be used to perform audits and surveillances provided the LES QA Director or QA Manager confirms and documents applicable QAPD requirements have been met and the individual has been certified in accordance with the QA procedure on auditor qualification and certification.

SECTION 19 PROVISIONS FOR CHANGE

This QAPD is reviewed and revised as necessary to reflect any changes that occur during the design, construction, operation, including maintenance and modifications, and decommissioning phases. In addition, this QAPD is revised when corrective actions, regulatory, organizational, or work scope changes warrant changes to the LES QA Program. The LES QAPD is maintained current through design, construction, operation and decommissioning of the facility. The LES QAPD is kept current as the design, construction, operation, and decommissioning activities progress, and appropriate changes are made based on any of the following:

- Lessons learned from audit and assessment findings,
- Program improvements identified from analysis of trends, and
- Changes due to regulations, commitments, reorganizations, revised project schedule, or program improvements from continuous review of assessment results and process improvement initiatives.

Changes to the LES QA Program shall be incorporated in this QAPD and submitted to the NRC within 30 days of implementation prior to and after NRC issuance of the License. Any changes that reduce commitments in the approved QAPD, including those commitments that address the safety program and integrated safety analysis regulatory requirements, as well as the QA Level requirements in this QAPD, will be submitted to the NRC for review and approval prior to implementation.

SECTION 20 QUALITY ASSURANCE PROGRAM FOR QA LEVEL 2 ACTIVITIES

This section outlines the owner defined Quality Assurance Program for QA Level 2 activities. For contractors, the QA Level 2 program shall be described in documents that must be approved by LES. The QA Level 2 program shall be applied to owner designated structures, systems, components, and activities. An International Organization for Standardization (ISO) 9000 series QA program is acceptable for QA Level 2 applications provided it complies with LES QAPD requirements and the ISO program is reviewed and approved by the LES QA Director.

Requirements for QA Level 2 are defined below. QA Level 2 requirements shall not be applied to IROFS or items that may affect the functions of the IROFS.

ORGANIZATION

The organization, lines of responsibility and authority are clearly established and documented.

PERSONNEL QUALIFICATIONS

Measures are established to provide for indoctrination and training of personnel to ensure suitable proficiency is achieved and maintained. Where specific qualifications are required by codes and standards, measures shall be taken to document the qualifications.

PROCEDURES

Work activities are performed in accordance with written procedures. Procedures shall contain the appropriate criteria for determining that prescribed activities have been satisfactorily accomplished.

DOCUMENT CONTROL

Procedures are established to ensure that appropriate documents are properly initiated, changed, and controlled to prevent use of incorrect or superseded documents.

DESIGN CONTROL

The design shall be defined, controlled, and verified. Applicable design inputs shall be appropriately specified on a timely basis and correctly translated into design documents. Design interfaces are identified and controlled. Design adequacy is verified by persons independent of those who performed the design. Design changes are governed by control measures commensurate with those applied to the original design. Design of systems, structures or components may be verified by the development and service testing of hardware similar to the equipment to be used in the facility. Installation and use of this type of equipment requires approval of LES management.

CONTROL OF PURCHASED ITEMS AND SERVICES

Measures are established to ensure conformance with the specified requirements. Measures are established to ensure suppliers of materials, equipment, or services are capable of supplying these items to the quality specified in the procurement documents. This may be done by evaluation and approval of the supplier's products and facilities or audits of the supplier's quality program.

CONTROL OF PROCESSES, MEASURING AND TEST EQUIPMENT

Processes affecting quality of items or services are controlled. Special processes such as welding, heat treating, and nondestructive examination shall be performed by certified personnel using certified procedures in accordance with specified requirements. To maintain accuracy within specified limits, the LES QA Program requires that devices (e.g., tools, gauges, instruments), and measuring and test equipment including process-related instrumentation and controls that are used in activities affecting the quality of items, are properly controlled, calibrated, and adjusted at specified periods in accordance with written procedures.

INSPECTIONS

Inspections required to verify conformance of an item or activity to specified requirements are planned and executed. Characteristics to be inspected and inspection methods to be employed are specified. Inspection results are documented. Inspections for acceptance are performed by persons other than those who performed the work being inspected.

NONCONFORMANCES AND CORRECTIVE ACTION

Measures are established so conditions adverse to required quality are promptly identified and corrected. Controls are established to prevent inadvertent installation or use of items that do not conform to specified requirements.

RECORDS

Records that furnish documentary evidence of quality are specified, prepared, and maintained. Records shall be legible, identifiable, and retrievable. Records are protected against damage, deterioration, and loss. Requirements and responsibilities for record transmittal, distribution, retention, maintenance, and disposition are established and documented.

AUDITS AND ASSESSMENTS

Measures are established to verify compliance with the LES QA Program and to determine its effectiveness. The results are documented and reported to and reviewed by responsible management. Follow-up action shall be taken where indicated.

FIGURES

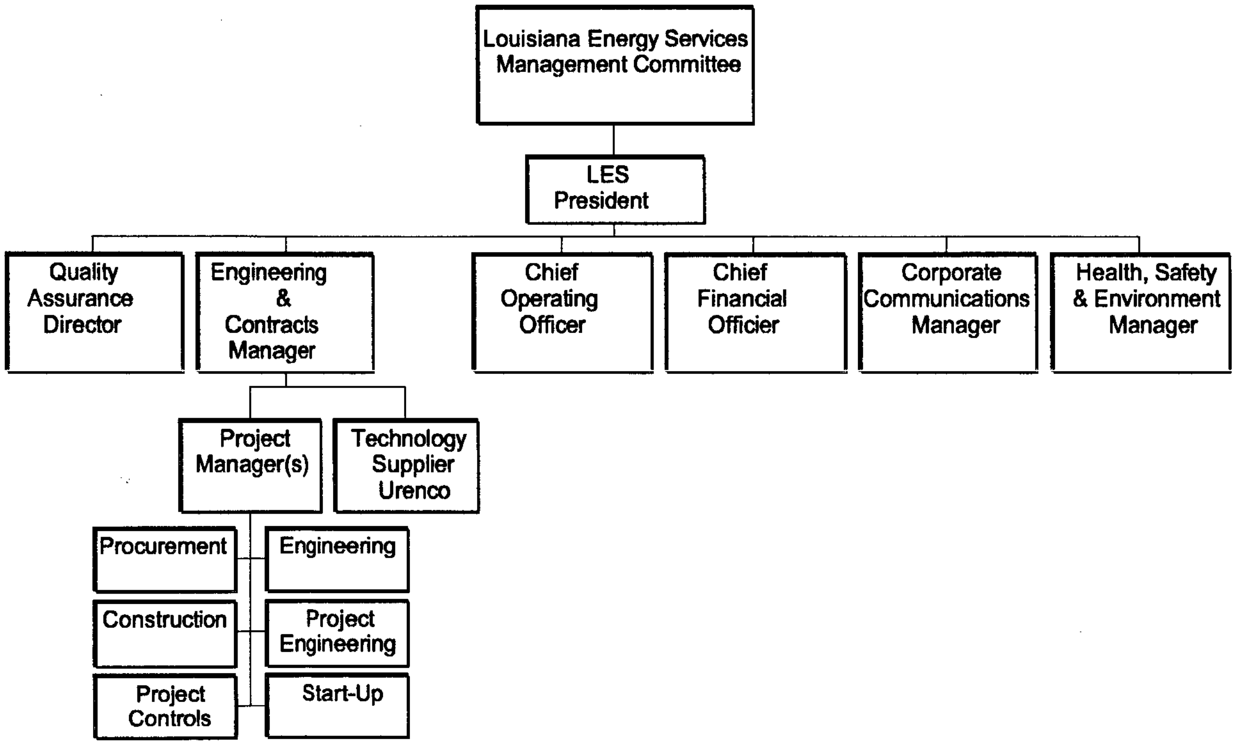

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Figure A1.dwg

FIGURE A1
LES CORPORATE, DESIGN AND CONSTRUCTION ORGANIZATION

REVISION 4 APRIL 2005

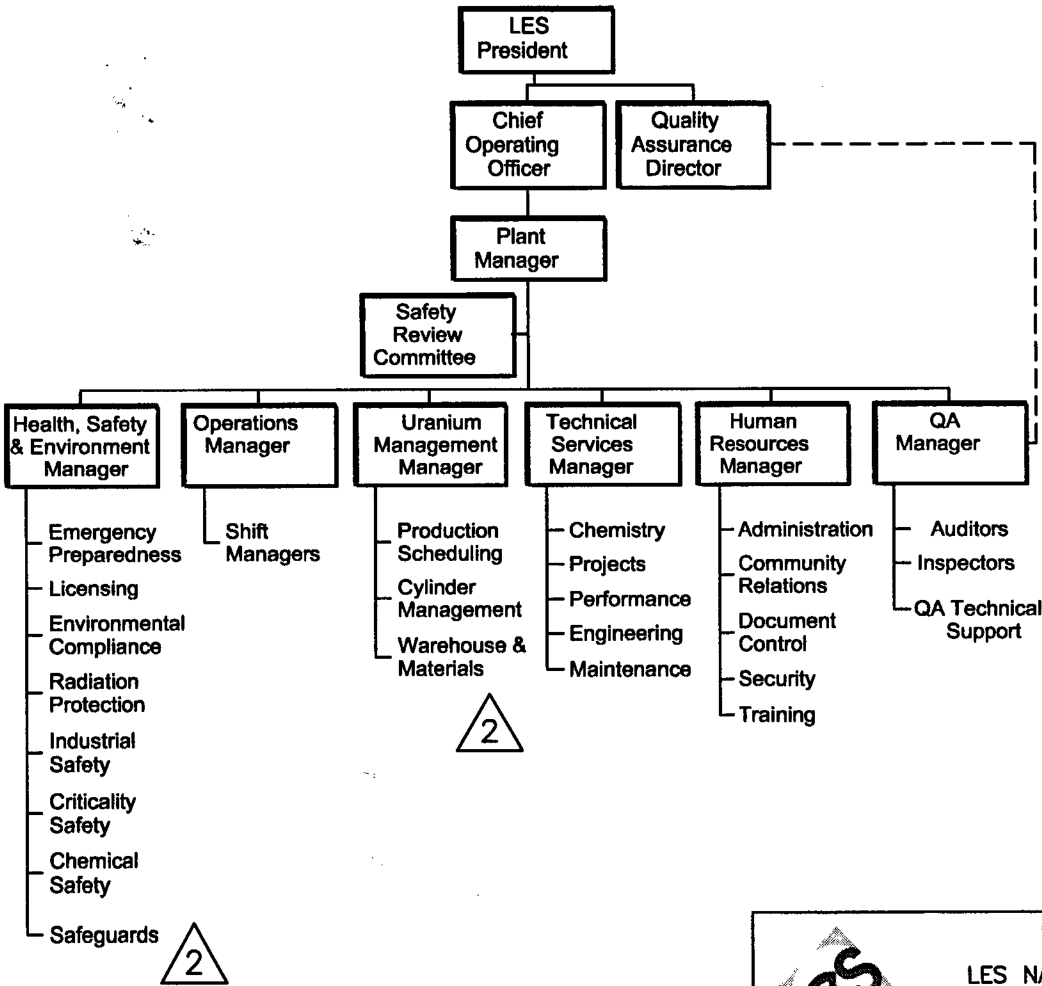

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Figure A2.dwg

FIGURE A2

LES NATIONAL ENRICHMENT PLAN
OPERATING ORGANIZATION

REVISION 2 DATE: JULY 2004

NATIONAL
ENRICHMENT
FACILITY

INTEGRATED SAFETY ANALYSIS SUMMARY

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1.0 PURPOSE

The purpose of this document, the National Enrichment Facility (NEF) Integrated Safety Analysis (ISA) Summary, is to provide a synopsis of the results of the NEF ISA, including the information specified in 10 CFR 70.65(b) (CFR, 2003a). An ISA identifies potential accident sequences in facility operations, designates items relied on for safety (IROFS) to either prevent such accidents or mitigate their consequences to an acceptable level, and describes management measures to provide reasonable assurance of the availability and reliability of IROFS. The NEF ISA Summary principally differs from the NEF ISA by focusing on higher risk accident sequences with consequences that could exceed the performance criteria of 10 CFR 70.61 (CFR, 2003b).

1.0.1 References

CFR, 2003a. Title 10, Code of Federal Regulations, Section 70.65, Additional content of applications, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Section 70.61, Performance requirements, 2003.

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2.0 SCOPE

The following information, as a minimum, is included in the National Enrichment Facility (NEF) Integrated Safety Analysis (ISA) Summary.

1. A general description of the site with emphasis on those factors that could affect safety (e.g., meteorology, seismology).
2. A general description of the facility with emphasis on those areas that could affect safety, including an identification of the controlled area boundaries.
3. A description of each process analyzed in the ISA, the hazards that were identified in the ISA, and a general description of the types of accident sequences.
4. Information that demonstrates compliance with the performance requirements of 10 CFR 70.61 (CFR, 2003a), including a description of the management measures, the requirements for criticality monitoring and alarms in 10 CFR 70.24 (CFR, 2003b), and the requirements of 10 CFR 70.64 (CFR, 2003c).
5. A description of the team, qualifications, and the methods used to perform the ISA.
6. A list briefly describing each item relied on for safety in sufficient detail to understand their functions in relation to the performance requirements of 10 CFR 70.61 (CFR, 2003a).
7. A description of the proposed quantitative standards used to assess the consequences to an individual from acute chemical exposure to licensed material or chemicals produced from licensed materials which are on-site, or expected to be on-site.
8. A descriptive list that identifies all items relied on for safety that are the sole item preventing or mitigating an accident sequence that exceeds the performance requirements of 10 CFR 70.61 (CFR, 2003a).
9. A description of the definitions of unlikely, highly unlikely, and credible as used in the evaluations in the ISA.

2.0.1 References

CFR, 2003a. Title 10, Code of Federal Regulations, Section 70.61, Performance requirements, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Section 70.24, Criticality accident requirements, 2003.

CFR, 2003c. Title 10, Code of Federal Regulations, Section 70.64, Requirements for new facilities or new processes at existing facilities, 2003.

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3.0 APPLICABLE REGULATORY REQUIREMENTS / GUIDANCE

The requirement to prepare and submit an Integrated Safety Analysis (ISA) Summary for Nuclear Regulatory Commission (NRC) approval is stated in 10 CFR 70.65(b) (CFR, 2003a). 10 CFR 70.65(b) (CFR, 2003a) also describes the contents of an ISA Summary. The ISA Summary has been developed following the guidance of NUREG-1520 (NRC, 2002) which meets the format, structure, and content of an ISA Summary that is consistent with the requirements of 10 CFR 70 (CFR, 2003b).

The information provided in the ISA Summary, the corresponding regulatory requirement, and the section of NUREG-1520 (NRC, 2002), Chapter 3 in which the NRC expectations for such information are presented are summarized below.

Information Category and Requirement	**10 CFR 70 Citation**	**NUREG-1520 Chapter 3 Reference**
Section 3.1 General Information		
• ISA methodology description	70.65(b)(5)	3.4.3.2(5)
• ISA Team description	70.65(b)(5)	3.4.3.2(5)
• Quantitative standards for acute chemical exposures	70.65(b)(7)	3.4.3.2(7)
• Definition of terms	70.65(b)(9)	3.4.3.2(9)
• Compliance with baseline design criteria and criticality monitoring and alarms	70.64 & 70.65(b)(4)	3.4.3.2(4D) 3.4.3.2(4C)
• Safety Program commitments	70.62(a)	3.4.3.1
Section 3.2 Site Description		
• Site description	70.65(b)(1)	3.4.3.2(1)
Section 3.3 Facility Description		
• Facility and Major Civil Structural Descriptions	70.65(b)(2)	3.4.3.2(2)
Section 3.4 Enrichment and Other Process Descriptions		
• Description of processes analyzed	70.65(b)(3)	3.4.3.2(3)
Section 3.5 Utility and Support Systems		
• Description of support systems analyzed	70.65(b)(3)	3.4.3.2(3)
Section 3.6 Process Hazards		
• Identification of hazards	70.65(b)(3)	3.4.3.2(3)
Section 3.7 Accident Sequences		
• General types of accident sequences	70.65(b)(3)	3.4.3.2(3)
• Risk ranking	70.65(b)(3)	3.4.3.2(3)
• Characterization of intermediate and high-risk accident sequences	70.65(b)(3)	3.4.3.2(3)
Section 3.8 Items Relied on For Safety (IROFS)		
• List and descriptions of IROFS at the system level	70.65(b)(6)	3.4.3.2(6)
• IROFS management measures	70.65(b)(4)	3.4.3.2(4B) 3.4.3.2(6)
• Sole IROFS	70.65(b)(8)	3.4.3.2(8)

3.0.1 References

CFR, 2003a. Title 10, Code of Federal Regulations, Section 70.65, Additional content of applications, 2003.

CFR, 2003b. Title 10, Code of Federal Regulations, Part 70, Domestic Licensing of Special Nuclear Material, 2003.

NRC, 2002. Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, NUREG-1520, U.S. Nuclear Regulatory Commission, March 2002.

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3.1 GENERAL INTEGRATED SAFETY ANALYSIS (ISA) INFORMATION

3.1.1 ISA Methods

This section outlines the approach utilized for performing the integrated safety analysis (ISA) of the process accident sequences. The approach used for performing the ISA is consistent with Example Procedure for Accident Sequence Evaluation, Appendix A to Chapter 3 of NUREG-1520 (NRC, 2002). This approach employs a semi-quantitative risk index method for categorizing accident sequences in terms of their likelihood of occurrence and their consequences of concern. The risk index method framework identifies which accident sequences have consequences that could exceed the performance requirements of 10 CFR 70.61 (CFR, 2003c) and, therefore, require designation of items relied on for safety (IROFS) and supporting management measures. Descriptions of these general types of higher consequence accident sequences are reported in the ISA Summary.

The ISA is a systematic analysis to identify plant and external hazards and the potential for initiating accident sequences, the potential accident sequences, the likelihood and consequences, and the IROFS.

The ISA uses a hazard analysis method to identify the hazards which are relevant for each system or facility. The ISA Team reviewed the hazard identified for the "credible worst-case" consequences. All credible high or intermediate severity consequence accident scenarios were assigned accident sequence identifiers, accident sequence descriptions, and a risk index determination was made.

The risk index method is regarded as a screening method, not as a definitive method of proving the adequacy or inadequacy of the IROFS for any particular accident.

The tabular accident summary resulting from the ISA identifies, for each sequence, which engineered or administrative IROFS must fail to allow the occurrence of consequences that exceed the levels identified in 10 CFR 70.61 (CFR, 2003c).

For this license application, two ISA Teams were formed. This was necessary because the sensitive nature of some of the facility design information related to the enrichment process required the use of personnel with the appropriate national security clearances. This team performed the ISA on the Cascade System, Contingency Dump System, Centrifuge Test System and the Centrifuge Post Mortem System. This ISA Team is referred to as the Classified ISA Team. The Non-Classified Team, referred to in the remainder of this text as the ISA Team, performed the ISA on the remainder of the facility systems and structures. In addition, the (non-classified) ISA Team performed the External Events and Fire Hazard Assessment for the entire facility.

In preparing for the ISA, the Accident Analysis in the Safety Analysis Report (LES, 1993) for the Claiborne Enrichment Center was reviewed. In addition, experienced personnel with familiarity with the gas centrifuge enrichment technology safety analysis where used on the ISA Team. This provides a good peer check of the final ISA results.

A procedure was developed to guide the conduct of the ISA. This procedure was used by both teams. In addition, there were common participants on both teams to further integrate the approaches employed by both teams. These steps were taken to ensure the consistency of the results of the two teams. A non-classified summary of the results of the Classified ISA has been prepared and incorporated into the ISA Summary.

3.1.1.1 Hazard Identification

The hazard and operability (HAZOP) analysis method was used for identifying the hazards for the Uranium Hexafluoride (UF_6) process systems and Technical Services Building systems. This method is consistent with the guidance provided in NUREG-1513 (NRC, 2001) and NUREG-1520 (NRC, 2002). The hazards identification process results in identification of physical, radiological or chemical characteristics that have the potential for causing harm to site workers, the public, or to the environment. Hazards are identified through a systematic review process that entails the use of system descriptions, piping and instrumentation diagrams, process flow diagrams, plot plans, topographic maps, utility system drawings, and specifications of major process equipment. In addition, criticality hazards identification were performed for the areas of the facility where fissile material is expected to be present. The criticality safety analyses contain information about the location and geometry of the fissile material and other materials in the process, for both normal and credible abnormal conditions. The ISA input information is included in the ISA documentation and is available to be verified as part of an on-site review.

The hazard identification process documents materials that are:

- Radioactive
- Fissile
- Flammable
- Explosive
- Toxic
- Reactive.

The hazard identification also identifies potentially hazardous process conditions. Most hazards were assessed individually for the potential impact on the discrete components of the process systems. However, for hazards from fires (external to the process system) and external events (seismic, severe weather, etc.), the hazards were assessed on a facility wide basis.

For the purpose of evaluating the impacts of fire hazards, the ISA team considered the following:

- Postulated the development of a fire occurring in in-situ combustibles from an unidentified ignition source (e.g., electrical shorting, or other source)
- Postulated the development of a fire occurring in transient combustibles from an unidentified ignition source (e.g., electrical shorting, or other source)
- Evaluated the uranic content in the space and its configuration (e.g., UF_6 solid/gas in cylinders, UF_6 gas in piping, UF_6 and/or byproducts bound on chemical traps, Uranyl Fluoride (UO_2F_2) particulate on solid waste or in solution). The appropriate configuration

was considered relative to the likelihood of the target releasing its uranic content as a result of a fire in the area.

In order to assess the potential severity of a given fire and the resulting failures to critical systems, the facility Fire Hazard Analysis was consulted. However, since the design supporting the license submittal for this facility is not yet at the detailed design stage, detailed in-situ combustible loading and in-situ combustible configuration information is not yet available. Therefore, in order to place reasonable and conservative bounds on the fire scenarios analyzed, the ISA Team estimated in-situ combustible loadings based on information of the in-situ combustible loading from Urenco's Almelo SP-5 plant (on which the National Enrichment Facility (NEF) design is based). This information from SP-5 indicates that in-situ combustible loads are expected to be very low.

The Fire Safety Management Program will limit the allowable quantity of transient combustibles in critical plant areas (i.e., uranium areas). Nevertheless, the ISA Team still assumed the presence of moderate quantities of ordinary (Class A) combustibles (e.g., trash, packing materials, maintenance items or packaging, etc.) in excess of anticipated procedural limits. This was not considered a failure of the associated administrative IROFS feature for controlling/ minimizing transient combustible loading in all radiation/uranium areas. Failure of the IROFS is connoted as the presence of extreme or severe quantities of transients (e.g., large piles of combustible solids, bulk quantities of flammable/combustible liquids or gases, etc.). The Urenco ISA Team representatives all indicated that these types of transient combustible conditions do not occur in the European plants. Accordingly, and given the orientation and training that facility employees will receive indicating that these types of fire hazards are unacceptable, the administrative IROFS preventing severe accumulations has been assigned a high degree of reliability. Refer to Section 3.8.3 for additional discussion.

Fires that involve additional in-situ or transient combustibles from outside each respective fire area could result in exposure of additional uranic content being released in a fire beyond the quantities assumed above. For this reason, fire barriers are needed to ensure that fires cannot propagate from non-uranium containing areas into uranium (U) areas or from one U area to another U area (unless the uranium content in the space is insignificant, i.e., would be a low consequence event). Fire barriers shall be designed with adequate safety margin such that the total combustible loading (in-situ and transient) allowed to expose the barrier will not exceed 80% of the hourly fire resistance rating of the barrier.

For external events, the impacts were evaluated for the following hazards:

External events were considered at the site and facility level versus at individual system nodes. Specific external event HAZOP guidewords were developed for use during the external event portion of the ISA. The external event ISA considered both natural phenomena and man-made hazards. During the external event ISA team meeting, each area of the plant was discussed as to whether or not it could be adversely affected by the specific external event under consideration. If so, specific consequences were then discussed. If the consequences were known or assumed to be high, then a specific design basis with a likelihood of highly unlikely would be selected.

Given that external events were considered at the facility level, the ISA for external events was performed after the ISA team meetings for all plant systems were completed. This provided the best opportunity to perform the ISA at the site or facility level. Each external event was assessed for both the uncontrolled case and then for the controlled case. The controlled cases could be a specific design basis for that external event, IROFS or a combination of both. An Accident Sequence and Risk matrix was prepared for each external event.

External events evaluated included:

- Seismic
- Tornado, Tornado Missile and High Wind
- Snow and Ice
- Flooding
- Local Precipitation
- Other (Transportation and Nearby Facility Accidents)
- Aircraft
- Pipelines
- Highway
- Other Nearby Facilities
- Railroad
- On-site Use of Natural Gas
- Internal Flooding from On-Site Above Ground Liquid Storage Tanks.

The ISA is intended to give assurance that the potential failures, hazards, accident sequences, scenarios, and IROFS have been investigated in an integrated fashion, so as to adequately consider common mode and common cause situations. Included in this integrated review is the identification of IROFS function that may be simultaneously beneficial and harmful with respect to different hazards, and interactions that might not have been considered in the previously completed sub-analyses. This review is intended to ensure that the designation of one IROFS does not negate the preventive or mitigation function of another IROFS. An integration checklist is used by the ISA Team as a guide to facilitate the integrated review process.

Some items that warrant special consideration during the integration process are:

- Common mode failures and common cause situations.
- Support system failures such as loss of electrical power or city water. Such failures can have a simultaneous effect on multiple systems.
- Divergent impacts of IROFS. Assurance must be provided that the negative impacts of an IROFS, if any, do not outweigh the positive impacts; i.e., to ensure that the application of an IROFS for one safety function does not degrade the defense-in-depth of an unrelated safety function.
- Other safety and mitigating factors that do not achieve the status of IROFS that could impact system performance.
- Identification of scenarios, events, or event sequences with multiple impacts, i.e. impacts on chemical safety, fire safety, criticality safety, and/or radiation safety. For example, a flood might cause both a loss of containment and moderation impacts.

- Potential interactions between processes, systems, areas, and buildings; any interdependence of systems, or potential transfer of energy or materials.
- Major hazards or events, which tend to be common cause situations leading to interactions between processes, systems, buildings, etc.

3.1.1.2 Process Hazard Analysis Method

As noted above, the HAZOP method was used to identify the process hazards. The HAZOP process hazard analysis (PHA) method is consistent with the guidance provided in NUREG-1513 (NRC, 2001). Implementation of the HAZOP method was accomplished by either validating the Urenco HAZOPs for the NEF design or performing a new HAZOP for systems where there were no existing HAZOPs. In general, new HAZOPs were performed for the Technical Services Building (TSB) systems. In cases for which there was an existing HAZOP, the ISA Team, through the validation process, developed a new HAZOP.

For the UF_6 process systems, this portion of the ISA was a validation of the HAZOPs provided by Urenco. The validation process involved workshop meetings with the ISA Team. In the workshop meeting, the ISA Team challenged the results of the Urenco HAZOPs. As necessary the HAZOPs were revised/updated to be consistent with the requirements identified in 10 CFR 70 (CFR, 2003b) and as further described in NUREG-1513 (NRC, 2001) and NUREG-1520 (NRC, 2002).

To validate the Urenco HAZOPs, the ISA Team performed the following tasks:

- The Urenco process engineer described the salient points of the process system covered by the HAZOP being validated.
- The ISA Team divided the process "Nodes" into reasonable functional blocks.
- The process engineer described the salient points of the items covered by the "Node" being reviewed.
- The ISA Team reviewed the "Guideword" used in the Urenco HAZOP to determine if the HAZOP is likely to identify all credible hazards. A representative list of the guidewords used by the ISA Team is provided in Table 3.1-1, HAZOP Guidewords, to ensure that a complete assessment was performed.
- The ISA Team Leader introduced each Guideword being considered in the ISA HAZOP and the team reviewed and considered the potential hazards.
- For each potential hazard, the ISA Team considered the causes, including potential interactions among materials. Then, for each cause, the ISA Team considered the consequences and consequence severity category for the consequences of interest (Criticality Events, Chemical Releases, Radiation Exposure, Environment impacts). A statement of "No Safety Issue" was noted in the system HAZOP table for consequences of no interest such as maintenance problems or industrial personnel accidents.
- For each hazard, the ISA Team considered existing safeguards designed to prevent the hazard from occurring.
- For each hazard, the ISA Team also considered any existing design features that could mitigate/reduce the consequences.

- The Urenco HAZOP was modified to reflect the ISA Team's input in the areas of hazards, causes, consequences, safeguards and mitigating features.
- For each external event hazard, the ISA Team determined if the external hazard is credible (i.e., external event initiating frequency >10^{-6} per year).
- When all of the Guidewords had been considered for a particular node, the ISA Team applied the same process and guidewords to the next node until the entire process system was completed.

The same process as above was followed for the TSB systems, except that instead of using the validation process, the ISA Team developed a completely new HAZOP. This HAZOP was then used as the hazard identification input into the remainder of the process.

The results of the ISA Team workshops are summarized in the ISA HAZOP Table, which forms the basis of the hazards portion of the Hazard and Risk Determination Analysis. The HAZOP tables are contained in the ISA documentation. The format for this table, which has spaces for describing the node under consideration and the date of the workshop, is provided in Table 3.1-2, ISA HAZOP Table Sample Format. This table is divided into 7 columns:

GUIDEWORD	Identifies the Guideword under consideration.
HAZARD	Identifies any issues that are raised.
CAUSES	Lists any and all causes of the hazard noted.
CONSEQUENCES	Identifies the potential and worst case consequence and consequences severity category if the hazard goes uncontrolled.
SAFEGUARDS	Identifies the engineered and/or administrative protection designed to prevent the hazard from occurring.
MITIGATION	Identifies any protection, engineered or otherwise, that can mitigate/reduce the consequences.
COMMENTS	Notes any comments and any actions requiring resolution.

This approach was used for all of the process system hazard identifications. The "Fire" and "External Events" guidewords were handled as a facility-wide assessment and were not explicitly covered in each system hazard evaluation.

The results of the HAZOP are used directly as input to the risk matrix development.

3.1.1.3 Risk Matrix Development

3.1.1.3.1 Consequence Analysis Method

10 CFR 70.61 (CFR, 2003c) specifies two categories for accident sequence consequences: "high consequences" and "intermediate consequences." Implicitly there is a third category for accidents that produce consequences less than "intermediate." These are referred to as "low consequence" accident sequences. The primary purpose of PHA is to identify all uncontrolled and unmitigated accident sequences. These accident sequences are then categorized into one

of the three consequence categories (high, intermediate, low) based on their forecast radiological, chemical, and/or environmental impacts.

For evaluating the magnitude of the accident consequences, calculations were performed using the methodology described in the ISA documentation. Because the consequences of concern are the chemotoxic exposure to hydrogen fluoride (HF) and UO_2F_2, the dispersion methodology discussed in Section 6.3.2 was used. The dose consequences for all of the accident sequences were evaluated and compared to the criteria for "high" and "intermediate" consequences. The inventory of uranic material for each accident considered was dependent on the specific accident sequence. For criticality accidents, the consequences were conservatively assumed to be high for both the public and workers.

Table 3.1-3, Consequence Severity Categories Based on 10 CFR 70.61, presents the radiological and chemical consequence severity limits of 10 CFR 70.61 (CFR, 2003c) for each of the three accident consequence categories. Table 3.1-4, Chemical Dose Information, provides information on the chemical dose limits specific to the NEF.

3.1.1.3.2 Likelihood Evaluation Method

10 CFR 70.61 (CFR, 2003c) also specifies the permissible likelihood of occurrence of accident sequences of different consequences. "High consequence" accident sequences must be "highly unlikely" and "intermediate consequence" accident sequences must be "unlikely." Implicitly, accidents in the "low consequence" category can have a likelihood of occurrence less than "unlikely" or simply "not unlikely." Table 3.1-5, Likelihood Categories Based on 10 CFR 70.61, shows the likelihood of occurrence limits of 10 CFR 70.61 (CFR, 2003c) for each of the three likelihood categories.

The definitions of "not unlikely" and "unlikely" are taken from NUREG-1520 (NRC, 2002). The definition of "highly unlikely" is taken from NUREG-1520 (NRC, 2002). Additionally, a qualitative determination of "highly unlikely" can apply to passive design component features (e.g., tanks, piping, cylinders, etc.) of the facility that do not rely on human interface to perform the criticality safety function (i.e., termed "safe-by-design"). Safe-by-design components are those components that by their physical size or arrangement have been shown to have a $k_{eff} < 0.95$. The definition of safe-by-design components encompasses two different categories of components. The first category includes those components that are safe-by-volume, safe-by-diameter or safe-by-slab thickness. A set of generic conservative criticality calculations has determined the maximum volume, diameter, or slab thickness (i.e., safe value) that would result in a $k_{eff} < 0.95$. A component in this category has a volume, diameter or slab thickness that is less than the associated safe value resulting from the generic conservative criticality calculations and therefore the k_{eff} associated with this component is < 0.95. The components in the second category require a more detailed criticality analysis (i.e., a criticality analysis of the physical arrangement of the component's design configuration) to show that k_{eff} is < 0.95. In the second category of components, the design configuration is not bounded by the results of the generic conservative criticality calculations for maximum volume, diameter, or slab thickness that would result in a $k_{eff} < 0.95$. Examples of components in this second category are the product pumps that have volumes greater than the safe-by-volume value, but are shown by specific criticality analysis to have a $k_{eff} < 0.95$.

For failure of passive safe-by-design components to be considered "highly unlikely," these components must also meet the criterion that the only potential means to effect a change that might result in a failure to function, would be to implement a design change (i.e., geometry deformation as a result of a credible process deviation or event does not adversely impact the performance of the safety function). The evaluation of the potential to adversely impact the safety function of these passive design features includes consideration of potential mechanisms to cause bulging, corrosion, and breach of confinement/leakage and subsequent accumulation of material. The evaluation further includes consideration of adequate controls to ensure that the double contingency principle is met. For each of these passive design components, it must be concluded, that there is no credible means to effect a geometry change that might result in a failure of the safety function and that significant margin exists. For components that are safe-by-volume, safe-by-diameter, or safe-by-slab thickness (i.e., first category of safe-by-design components), significant margin is defined as a margin of at least 10%, during both normal and upset conditions, between the actual design parameter value of the component and the value of the corresponding critical design attribute. For components that require a more detailed criticality analysis (i.e., second category of safe-by-design components), significant margin is defined as $k_{eff} < 0.95$, where $k_{eff} = k_{calc} + 3\sigma_{calc}$. This margin is considered acceptable since the calculation of k_{eff} also conservatively assumes the components are full of uranic breakdown material at maximum enrichment, the worst credible moderation conditions exist, and the worst credible reflection conditions exist.

The demonstration of significant margin to meet "highly unlikely" is provided, for each of the components listed in Tables 3.7-6 through 3.7-21, in the following classified documents.

- ETC4009554, Criticality Assessment of Passive Safe-by-Design Components, Decontamination Workshop
- ETC4009555, Criticality Assessment of Passive Safe-by-Design Components, Mass Spectrometry Laboratory
- ETC4009556, Criticality Assessment of Passive Safe-by-Design Components, Chemical Laboratory System
- ETC4009557, Criticality Assessment of Passive Safe-by-Design Components, Fomblin Oil Recovery System
- ETC4009558, Criticality Assessment of Passive Safe-by-Design Components, Solid Waste Collection System
- ETC4009559, Criticality Assessment of Passive Safe-by-Design Components, Product Blending System
- ETC4009561, Criticality Assessment of Passive Safe-by-Design Components, Cascade System
- ETC4009565, Criticality Assessment of Passive Safe-by-Design Components, Centrifuge Test System
- ETC4009566, Criticality Assessment of Passive Safe-by-Design Components, Centrifuge Post Mortem Facility
- ETC4009567, Criticality Assessment of Passive Safe-by-Design Components, Contingency Dump System
- ETC4009609, Criticality Assessment of Passive Safe-by-Design Components, Tails System

- ETC4009614, Criticality Assessment of Passive Safe-by-Design Components, Product System
- ETC4009677, Criticality Assessment of Passive Safe-by-Design Components, Liquid Effluent Collection and Treatment System
- ETC4009679, Criticality Assessment of Passive Safe-by-Design Components, Ventilated Room System
- ETC4009723, Criticality Assessment of Passive Safe-by-Design Components, Cylinder Preparation System
- ETC4009730, Criticality Assessment of Passive Safe-by-Design Components, Liquid Sampling System

These classified documents are incorporated by reference into this ISA Summary.

In addition, the configuration management system required by 10 CFR 70.72 (implemented by the NEF Configuration Management Program) ensures the maintenance of the safety function of these features and assures compliance with the double contingency principle, as well as the defense-in-depth criterion of 10 CFR 70.64(b).

The definition of "not credible" is also taken from NUREG-1520 (NRC, 2002). If an event is not credible, IROFS are not required to prevent or mitigate the event. The fact that an event is not "credible" must not depend on any facility feature that could credibly fail to function. One cannot claim that a process does not need IROFS because it is "not credible" due to characteristics provided by IROFS. The implication of "credible" in 10 CFR 70.61 (CFR, 2003c) is that events that are not "credible" may be neglected.

Any one of the following independent acceptable sets of qualities could define an event as not credible:

a. An external event for which the frequency of occurrence can conservatively be estimated as less than once in a million years

b. A process deviation that consists of a sequence of many unlikely human actions or errors for which there is no reason or motive (In determining that there is no reason for such actions, a wide range of possible motives, short of intent to cause harm, must be considered. Necessarily, no such sequence of events can ever have actually happened in any fuel cycle facility.)

c. Process deviations for which there is a convincing argument, given physical laws that they are not possible, or are unquestionably extremely unlikely.

3.1.1.3.3 Risk Matrix

The three categories of consequence and likelihood can be displayed as a 3 x 3 risk index matrix. By assigning a number to each category of consequence and likelihood, a qualitative risk index can be calculated for each combination of consequence and likelihood. The risk index equals the product of the integers assigned to the respective consequence and likelihood categories. The risk index matrix, along with computed risk index values, is illustrated in Table 3.1-6, Risk Matrix with Risk Index Values. The shaded blocks identify accidents of which the consequences and likelihoods yield an unacceptable risk index and for which IROFS must be applied.

The risk indices can initially be used to examine whether the consequences of an uncontrolled and unmitigated accident sequence (i.e., without any IROFS) could exceed the performance requirements of 10 CFR 70.61 (CFR, 2003c). If the performance requirements could be exceeded, IROFS are designated to prevent the accident or to mitigate its consequences to an acceptable level. A risk index value less than or equal to four means the accident sequence is acceptably protected and/or mitigated. If the risk index of an uncontrolled and unmitigated accident sequence exceeds four, the likelihood of the accident must be reduced through designation of IROFS. In this risk index method, the likelihood index for the uncontrolled and unmitigated accident sequence is adjusted by adding a score corresponding to the type and number of IROFS that have been designated.

3.1.1.4 Risk Index Evaluation Summary

The results of the ISA are summarized in tabular form (see Section 3.7, General Types of Accident Sequences). This table includes the accident sequences identified for this facility. The accident sequences were not grouped as a single accident type but instead were listed individually in the table. The Table has columns for the initiating event and for IROFS. IROFS may be mitigative or preventive. Mitigative IROFS are measures that reduce the consequences of an accident. The phrase "uncontrolled and/or unmitigated consequences" describes the results when the system of existing preventive IROFS fails and existing mitigation also fails. Mitigated consequences result when the preventive IROFS fail, but mitigative measures succeed. Index numbers are assigned to initiating events, IROFS failure events, and mitigation failure events, based on the reliability characteristics of these items.

With redundant IROFS and in certain other cases, there are sequences in which an initiating event places the system in a vulnerable state. While the system is in this vulnerable state, an IROFS must fail for the accident to result. Thus, the frequency of the accident depends on the frequency of the first event, the duration of vulnerability, and the frequency of the second IROFS failure. For this reason, the duration of the vulnerable state is considered, and a duration index is assigned. The values of all index numbers for a sequence, depending on the number of events involved, are added to obtain a total likelihood index, T. Accident sequences are then assigned to one of the three likelihood categories of the risk matrix, depending on the value of this index in accordance with Table 3.1-8, Determination of Likelihood Category.

The values of index numbers in accident sequences are assigned considering the criteria in Tables 3.1-9 through 3.1-11. Each table applies to a different type of event. Table 3.1-9, Failure Frequency Index Numbers, applies to events that have *frequencies* of occurrence, such as initiating events and certain IROFS failures. In addition to further support the failure frequency index numbers used in the ISA (i.e., when ISA Summary Tables 3.7-2 and 3.7-4 state "This failure frequency index was selected based on evidence from history of similarly designed Urenco European plant…"), operating data from similar systems, components, and safety functions at the Urenco Almelo SP5 facility, which is similar to the NEF design, is reviewed. This review is conducted using searches of computer-based databases at the Urenco Almelo facility. A list of ISA Summary initiating events caused by component failures or human events is developed. Using this list of initiating events, keyword searches of computer based databases for plant control systems, operational logs, and maintenance records are performed. The resulting information relevant to the Almelo SP5 facility is extracted for further review, evaluation, and comparison to the failure frequency index number(s) used in the applicable ISA Summary accident sequences. When failure *probabilities* are required for an event, Table 3.1-10, Failure Probability Index Numbers, provides the index values. Table 3.1-11, Failure Duration Index Numbers, provides index numbers *for durations* of failure. These are used in certain accident sequences where two IROFS must simultaneously be in a failed state. In this case, one of the two controlled parameters will fail first. It is then necessary to consider the duration that the system remains vulnerable to failure of the second. This period of vulnerability can be terminated in several ways. The first failure may be "fail-safe" or be continuously monitored, thus alerting the operator when it fails so that the system may be quickly placed in a safe state. Or the IROFS may be subject to periodic surveillance tests for hidden failures. When hidden failures are possible, these surveillance intervals limit the duration that the system is in a vulnerable state. The reverse sequences, where the second IROFS fails first, should be considered as a separate accident sequence. This is necessary because the failure frequency and the duration of outage of the first and the second IROFS may differ. The values of these duration indices are not merely judgmental. They are directly related to the time intervals used for surveillance and the time needed to render the system safe.

The duration of failure is accounted for in establishing the overall likelihood that an accident sequence will continue to the defined consequence. Thus, the time to discover and repair the failure is accounted for in establishing the risk of the postulated accident.

The total likelihood index is the sum of the indices for all the events in the sequence, including those for duration. Consequences are assigned to one of the three consequence categories of the risk matrix, based on calculations or estimates of the actual consequences of the accident sequence. The consequence categories are based on the levels identified in 10 CFR 70.61 (CFR, 2003c). Multiple types of consequences can result from the same event. The consequence category is chosen for the most severe consequence.

In summarizing the ISA results, Table 3.7-1, Accident Sequence and Risk Index, provides two risk indices for each accident sequence to permit evaluation of the risk significance of the IROFS involved. To measure whether an IROFS has high risk significance, the table provides an "uncontrolled risk index," determined by modeling the sequence with all IROFS as failed (i.e., not contributing to a lower likelihood). In addition, a "controlled risk index" is also calculated, taking credit for the low likelihood and duration of IROFS failures. When an accident sequence has an uncontrolled risk index exceeding four but a controlled risk index of less than four, the IROFS involved have a high risk significance because they are relied on to achieve acceptable safety performance. Thus, use of these indices permits evaluation of the possible benefit of improving IROFS and also whether a relaxation may be acceptable.

3.1.2 ISA Team

There were two ISA Teams that were employed in the ISA. The first team worked on the non-classified portions of the facility and is referred to in the text as the ISA Team. The second team, referred to as the Classified ISA Team, performed the ISA on the classified elements of the facility. Both teams were selected with credentials consistent with the requirements in 10 CFR 70.65 (CFR, 2003a) and the guidance provided in NUREG-1520 (NRC, 2002). To facilitate consistency of results, common membership was dictated as demonstrated below (i.e., some members of the Non-Classified Team participated on the Classified Team. One of the members of the Classified Team participated in the ISA Team Leader Training, which was conducted prior to initiating the ISA. In addition, the Classified ISA Team Leader observed some of the non-classified ISA Team meetings.

The ISA was performed by a team with expertise in engineering, safety analysis and enrichment process operations. The team included personnel with experience and knowledge specific to each process or system being evaluated. The team was comprised of individuals who have experience, individually or collectively, in:

- Nuclear criticality safety
- Radiological safety
- Fire safety
- Chemical process safety
- Operations and maintenance
- ISA methods.

The ISA team leader was trained and knowledgeable in the ISA method(s) chosen for the hazard and accidents evaluations. Collectively, the team had an understanding of all process operations and hazards under evaluation.

The ISA Manager was responsible for the overall direction of the ISA. The process expertise was provided by the Urenco personnel on the team. In addition, the Team Leader has an adequate understanding of the process operations and hazards evaluated in the ISA, but is not the responsible cognizant engineer or enrichment process expert.

A description of the ISA Team, their areas of expertise, qualifications and experience is provided below.

ISA Team Member	Experience and Qualifications
Michael Kennedy, ISA Manager and Team Leader	Over 29 years experience in nuclear safety analyses and risk assessment. Advanced degrees in Nuclear Engineering. Completed ISA Team Leader training course.
Richard Turcotte, Team Leader	Over 25 years experience providing engineering and risk assessment support for nuclear plants. Significant experience in probabilistic risk assessment. Degreed Mechanical Engineer. Completed ISA Team Leader training course.

ISA Team Member	Experience and Qualifications
Melvin Gmyrek, Team Leader	Over 30 years experience in nuclear facility operations. Has held a number of reactor operator licenses and held positions as Senior Reactor Operator, shift supervisor and operations manager. Completed ISA Team Leader training course.
David Pepe, Scribe	Over 26 years experience in providing engineering and risk assessment support on nuclear facilities. Significant experience in probabilistic risk assessment. Degreed Nuclear Engineer. Completed ISA Team Leader training course.
Scott Tyler, Chemical/Fire Safety	Over 17 years experience in fire and chemical safety on nuclear and non-nuclear facilities. Experienced in process hazard and consequence analysis. Degreed engineer in Fire Protection and Safety Engineering Technology and a registered Professional Fire Protection Engineer.
Richard Dible, Fire Safety	Over 19 years experience in fire protection and analysis. Degreed engineer in Fire Protection and Safety Engineering.
Douglas Setzer, Chemical/Fire Safety	Over 16 years experience in design and analysis in chemical and fire safety. Experienced in process hazard and consequence analysis. Degreed engineer in Mechanical and Chemical engineering. Registered Professional Fire Protection Engineer.
Kevin Morrissey, Criticality Safety	Over 24 years of nuclear industry experience, including particle transport methods, nuclear criticality, activation analysis and reactor physics.
Mark Strum, Radiological Safety	Over 30 years of nuclear utility experience performing radiological assessments supporting the design, licensing and operation of both PWR and BWR nuclear power plant facilities. Degreed nuclear engineer with an advanced degree in Radiological Sciences and Protection.
Chris Andrews, Process Expert	Over 30 years experience in the licensing, engineering and safety analysis of gas centrifuge enrichment technology. Senior Manager responsible for safety analysis and licensing for Urenco. Degree in Physics. Professional Engineer. Completed ISA Team Leader training course.
Allan Brown, Process Expert	Over 26 years experience in the design, operations, start-up, decommissioning of gas centrifuge enrichment facilities. Design Manager

ISA Team Member	Experience and Qualifications
	with responsibility for the NEF for Urenco. Degree in Physics.
Jan Kleissen, Operations Expert	Over 30 years experience in the operation and start-up of gas centrifuge enrichment plants. Production Manager at the Almelo SP-5 plant. The NEF is based on the SP-5 design. Degreed engineer.
Edwin Mulder, Operations Expert	Over four years experience in operations of gas centrifuge enrichment plant.
Herald Voschezang, Operations Expert	Over 19 years of experience with Urenco, predominantly in operations of gas centrifuge enrichment plants. Commissioning Manager of the Almelo SP-5 plant. The NEF is based on the SP-5 design. Degreed engineer.
Randy Campbell, Facility Engineering	Over 25 years experience in engineering, design and construction in the power (nuclear and fossil), chemicals, automotive and other various industries and 12 years nuclear experience. Degreed Mechanical Engineer.

Classified ISA Team Member	Experience and Qualifications
Andrew Pilkington, Team Leader/Risk Analysis	Over 14 years experience in nuclear and non-nuclear facility risk assessment. Significant experience in the risk assessment of gas centrifuge enrichment facilities. Knowledgeable in the HAZOP methodology. Degreed engineer.
Tony Duff, Scribe/Risk Analysis	Over 13 years experience in nuclear facility risk assessment. Most recent experience in gas centrifuge enrichment facility risk assessment. Degree in Applied Physics.
Chris Andrews, Process Safety	Over 30 years experience in the licensing, engineering and safety analysis of gas centrifuge enrichment technology. Senior Manager responsible for safety analysis and licensing for Urenco. Degree in Physics. Professional Engineer. Completed ISA Team Leader training course.
Edwin Mulder, Operations Expert	Over four years experience in operations of gas centrifuge enrichment plant.

Classified ISA Team Member	Experience and Qualifications
Philip Hale, Lead Engineer	Over 21 years experience in mechanical and process design engineering on gas centrifuge enrichment facilities. Lead design engineer for the NEF. Advanced degree in Mechanical Engineering.
Owen Parry, Criticality	Over 20 years experience in gas centrifuge technology. Most recent experience is in the criticality analysis related to gas centrifuge enrichment facilities. Degree in Chemistry and Doctoral degree in Physics.
Ian Forrest, Dump Systems	Over 27 years experience in design engineering. Presently package manager for work associated with development and qualification of Dump Systems, and providing related support for plant and projects. Degreed Mechanical Engineer.
Alan Coles, Fire Safety	Over 36 years experience in fire protection and fire safety.
Heather Tur, Test Facilities	Over 32 years experience in centrifuge research and development and centrifuge test facility operations.
Ian Crombie, Test Facilities	Over 20 years experience in design engineering related to gas centrifuge enrichment plant. Most recently involved in the NEF design.
Herald Voschezang, Operations Expert	Over 19 years of experience with Urenco, predominantly in operations of gas centrifuge enrichment plants. Commissioning Manager of the Almelo SP-5 plant. The NEF is based on the SP-5 design. Degreed engineer.
Stephen Thomas, Process Design Engineer	Over 25 years of experience. Approximately 10 years of centrifuge plant design experience. Design support for NEF design.

The management commitments related to the conduct and maintenance of the ISA are described in Section 3.1.8.2, Integrated Safety Analysis.

3.1.3 Selection of Quantitative Standards

Uranium hexafluoride (UF_6) is the only chemical of concern that will be used at the facility. For licensed material or hazardous chemicals produced from licensed materials, chemicals of concern are those that, in the event of release have the potential to exceed concentrations defined in 10 CFR Part 70 (CFR, 2003b). UF_6 represents a health hazard to facility workers and the public if released to atmosphere due to the radiological and toxicological properties of two byproducts – hydrogen fluoride (HF) and uranyl fluoride (UO_2F_2) – which are generated when UF_6 is released and reacts with water vapor in the air.

Criteria for evaluating potential releases and characterizing their consequences as either "high" or "intermediate" for members of the public and facility workers are presented in Table 3.1-3, Consequence Severity Categories Based on 10 CFR 70.61 and Table 3.1-4, Chemical Dose Information.

3.1.4 Hazards Analyzed

The hazards of concern for this facility are all related to either a loss of confinement (of UF_6) or criticality. All of the consequences of concern are the result of initiating events due to hazards that would result in accidents of these types. The initiating events considered for this facility are the result of failures in process components, human error or misoperation including maintenance activities, fires (external to the process), and external events (e.g., severe weather, seismic, transportation and industrial hazards). These initiating events or potential causes could result in a loss of enrichment system containment or criticality. In general, the loss of confinement would initially result in an in-leakage of air because the systems are at sub-atmospheric pressure. Moisture in the air would react with the UF_6 forming UO_2F_2 and HF as by-products. The HF, which would be in a gaseous form, could be transported through the facility and ultimately beyond the site boundary. HF is a toxic chemical with the potential to cause harm to the plant workers or the public.

A criticality event, if one should occur, is a potential source of damaging energy and would result in the release of prompt gamma rays and airborne fission products. The gamma rays and airborne fission products result in direct radiation and chemical/radiological inhalation dose exposure to plant workers and the public. Each portion of the plant, system, or component that may possibly contain enriched uranium is designed with criticality safety as an objective. Where there is a potential for significant in-process accumulations of enriched uranium, the plant design includes multiple features to minimize the possibilities for breakdown of criticality control features.

Nuclear criticality safety is evaluated for the design features of the plant system or component and for the operating practices that relate to maintaining criticality safety. The evaluation of individual systems or components and their interaction with other systems or components containing enriched uranium is performed to assure the criticality safety criteria are met. The nuclear criticality safety analyses provide a basis for the plant design and criticality hazards identifications performed as part of the ISA.

3.1.5 Criticality Monitoring and Alarms

The facility is provided with a Criticality Accident Alarm System (CAAS) as required by 10 CFR 70.24, Criticality accident requirements (CFR, 2003d). Areas where Special Nuclear Material (SNM) is handled, used, or stored in amounts at or above the 10 CFR 70.24 (CFR, 2003d) mass limits are provided with CAAS coverage.

The CAAS is designed, installed, and maintained in accordance with ANSI/ANS-8.3-1997 Criticality Accident Alarm System (ANSI, 1997) as modified by Regulatory Guide 3.71, Nuclear Criticality Safety Standards Fuels and Material Facilities (NRC, 1998).

CAAS coverage consists of an overlapping detection layout, where all required covered areas are monitored by a minimum of a pair (2) of gamma detectors. Detectors trip based on both steady radiation rate and time integrated total radiation dose levels. The detectors have a stated trigger response of 1mGy/hr (0.1 rad/hr) as a gamma radiation rate meter detector. Based on this design and the guidance provided in Appendix B of ANSI/ANS-8.3 (ANSI, 1997), the radius of detection must be less than 106 m (348 ft). Because of building steel spacing and equipment arrangement as well as a desire to maintain a factor of two safety margin, a radius of detection of 40 m (131 ft) is used in the design. This ensures that the CAAS is capable of detecting a criticality that produces an absorbed dose in soft tissue of 0.2 Gy (20 rads) of combined neutron and gamma radiation at an unshielded distance of 2 m (6.6 ft) from the reacting material within one minute. The CAAS will be uniform throughout the facility for the type of radiation detected, the mode of detection, the alarm signal, and the system dependability. The CAAS, if tripped, will automatically initiate a clearly audible signal in areas that must be evacuated.

The CAAS is provided with emergency power and is designed to remain operational during credible events or conditions, including fire, explosion, corrosive atmosphere, or seismic shock (equivalent to the site-specific design-basis earthquake or the equivalent value specified by the uniform building code).

Whenever the CAAS is not functional, compensatory measures, such as limiting access and restricting SNM movement, will be implemented. Should the CAAS coverage be lost and not restored within a specified number of hours, the operations will be rendered safe (by shutdown and quarantine) if necessary. Onsite guidance is provided and is based on process-specific considerations that consider applicable risk trade-off of the duration of reliance on compensatory measures versus the risk associated with process upset in shutdown.

3.1.6 Fire Hazards Analysis

Fire Hazards Analyses (FHAs) are conducted for the processing buildings located within the site boundary. The FHA evaluates the facility design with respect to fire safety codes, and ensures that the facility is designed and operated such that there is acceptable risk for postulated fire accident scenarios.

The results of the FHA have been used to identify potential fire initiators and accident sequences leading to radiological consequences or toxic chemical consequences. The FHA is a fundamental input for evaluating fire hazards in the ISA.

3.1.7 Baseline Design Criteria

10 CFR 70.64 (CFR, 2003e) specifies baseline design criteria (BDC) that must be used for new facilities. The ISA accident sequences for the credible high and intermediate consequence events for the NEF have defined the design basis events. The IROFS for these events and safety parameter limits ensure that the associated BDC are satisfied. IROFS safety parameter limits are available in the ISA documentation. These BDC have been used as bases for the design of the NEF.

A. Quality Standards and Records.

Structures, systems, and components (SSCs) that are determined to have safety significance are designed, fabricated, erected, and tested in accordance with the quality assurance criteria set forth in Appendix B to 10 CFR Part 50 (CFR, 2003f). Appropriate records of the design, fabrication, erection, procurement and testing of SSCs which are determined to have safety significance are maintained throughout the life of the facility. A safety function is a function performed by a SSC that prevents a release of UF_6 to the environment that could result in a dose to a member of the public of at least the limits provided in Section 3.1.3, Selection of Quantitative Standards. An SSC that performs a safety function is designated as an "item relied on for safety" (IROFS). Management Measures applicable to IROFS are discussed in Section 3.1.8.3, Management Measures.

B. Natural Phenomena Hazards.

Structures, systems, and components that are determined to have safety significance (IROFS) are designed to withstand the effects of, and be compatible with, the environmental conditions associated with operation, maintenance, shutdown, testing, and accidents for which the IROFS are required to function.

Natural phenomena hazards are identified in Section 3.2, Site Description.

C. Fire Protection.

Structures, systems, and components that are determined to have safety significance (IROFS) are designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Non-combustible and heat resistant materials are used wherever practical throughout the facility, particularly in locations vital to the control of hazardous materials and to the maintenance of safety control functions. IEEE-383 (ANSI/IEEE, 1974) fire resistant cabling shall be used for all uranic material system power, instrumentation and control circuits. Fire detection, alarm, and suppression systems are designed and provided with sufficient capacity and capability to minimize the adverse effects of fires and explosion on IROFS. The design includes provisions to protect against adverse effects that might result from either the operation or the failure of the fire suppression system.

D. Environmental and Dynamic Effects.

Structures, systems, and components that are determined to have safety significance (IROFS) are protected against dynamic effects, including effects of missiles and discharging fluids, that may result from natural phenomena, accidents at nearby industrial, military, or transportation facilities, equipment failure, and other similar events and conditions both inside and outside the facility.

E. Chemical Protection.

The design provides adequate protection against chemical risks produced from licensed material, facility conditions which affect the safety of licensed material, and hazardous chemicals produced from licensed material.

F. Emergency Capability.

Structures, systems, and components that are required to support the Emergency Plan are designed for emergencies. The design provides accessibility to the equipment of onsite and available offsite emergency facilities and services such as hospitals, fire and police departments, ambulance service, and other emergency agencies.

G. Utility Services.

Onsite utility service systems required to support IROFS shall be provided. Each utility service system required to support IROFS shall provide for the meeting of safety demands under normal and abnormal conditions.

Utility systems are described in Section 3.5, Utility and Support Systems.

H. Inspection, Testing, and Maintenance.

Structures, systems and components that are determined to have safety significance (IROFS) are designed to permit inspection, maintenance, and testing.

I. Criticality Control.

Safety Margins

The design of process and storage systems shall include demonstrable margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the process and storage conditions, in the data and methods used in calculations, and in the nature of the immediate environment under accident conditions. All process and storage systems should be designed and maintained with sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

Methods of Control

The major controlling parameters used in the facility are enrichment control, geometry control, moderation control and/or limitations on the mass as a function of enrichment.

Neutron Absorbers

Neutron absorbers are not needed and are not used at the NEF.

J. Instrumentation and Controls.

Instrumentation and control systems shall be provided to monitor variables and operating systems that are significant to safety over anticipated ranges for normal operation, for abnormal operation, for accident conditions, and for safe shutdown. These systems shall ensure adequate safety of process and utility service operations in connection with their safety function. The variables and systems that require constant surveillance and control include process systems having safety significance, the overall confinement system, confinement barriers and their associated systems, and other systems that affect the overall safety of the plant. Controls shall be provided to maintain these variables and systems within the prescribed operating ranges under all normal conditions. Instrumentation and control systems shall be designed to fail into a safe state or to assume a state demonstrated to be acceptable on some other basis if conditions such as disconnection, loss of energy or motive power, or adverse environments are experienced.

For hardware IROFS involving instrumentation that provides automatic prevention or mitigation of events, status and operation will be monitored by the plant control system (PCS) by means of an alarm. This alarm will be provided by an isolated, hardwired digital signal from the associated IROFS to the PCS programmable logic controller (PLC). This signal will only be directed from the associated IROFS to the PCS PLC. The required isolation is provided at the IROFS hardware interface in the process equipment for the connections to the PCS PLC. Consistent with IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations" (IEEE, 1971), the isolation devices will be classified as part of the IROFS boundary and will be designed such that no credible failure at the output of the isolation device shall prevent the associated IROFS from meeting its specified safety function.

K. Defense-in-Depth Practices.

The facility and system designs are based on defense-in-depth practices. The design incorporates a preference for engineered controls over administrative controls to increase overall system reliability. For criticality safety, the engineered controls preference is for use of passive engineered controls over active engineered controls. The design also incorporates features that enhance safety by reducing challenges to items relied on for safety. Facility and system IROFS are identified in Section 3.8, IROFS. The process systems are described in Section 3.4, Enrichment and Other Process Systems. The utility and support systems are described in Section 3.5, Utility and Support Systems. In addition to identifying the IROFS associated with each system, the system descriptions also identify the additional design and safety features (considerations) that provide defense-in-depth.

3.1.8 Safety Program Commitments

This section presents the commitments pertaining to the facility's safety program including the performance of an ISA. 10 CFR Part 70 (CFR, 2003b) contains a number of specific safety program requirements related to the integrated safety analysis (ISA). These include the primary requirements that an ISA be conducted, and that it evaluate and show that the facility complies with the performance requirements of 10 CFR 70.61 (CFR, 2003c).

The commitments for each of the three elements of the safety program defined in 10 CFR 70.62(a) (CFR, 2003g) are addressed below.

3.1.8.1 Process Safety Information

A. LES has compiled and maintains up-to-date documentation of process safety information. Written process-safety information is used in updating the ISA and in identifying and understanding the hazards associated with the processes. The compilation of written process-safety information includes information pertaining to:

   1. The hazards of all materials used or produced in the process, which includes information on chemical and physical properties such as are included on Material Safety Data Sheets meeting the requirements of 29 CFR 1910.1200(g) (CFR, 2003h).

   2. Technology of the process which includes block flow diagrams or simplified process flow diagrams, a brief outline of the process chemistry, safe upper and lower limits for controlled parameters (e.g., temperature, pressure, flow, and concentration), and evaluation of the health and safety consequences of process deviations.

3. Equipment used in the process including general information on topics such as the materials of construction, piping and instrumentation diagrams (P&IDs), ventilation, design codes and standards employed, material and energy balances, IROFS (e.g., interlocks, detection, or suppression systems), electrical classification, and relief system design and design basis.

The process-safety information described above is maintained up-to-date by the configuration management program.

B. LES has developed procedures and criteria for changing the ISA. This includes implementation of a facility change mechanism that meets the requirements of 10 CFR 70.72 (CFR, 2003i).

C. LES uses personnel with the appropriate experience and expertise in engineering and process operations to maintain the ISA. The ISA Team for the various processes consists of individuals who are knowledgeable in the ISA method(s) and the operation, hazards, and safety design criteria of the particular process.

The ISA Team for the initial ISA development is described in Section 3.1.2, ISA Team.

3.1.8.2 Integrated Safety Analysis

A. LES has conducted an ISA for each process, such that it identifies (i) radiological hazards, (ii) chemical hazards that could increase radiological risk, (iii) facility hazards that could increase radiological risk, (iv) potential accident sequences, (v) consequences and likelihood of each accident sequence and (vi) IROFS including the assumptions and conditions under which they support compliance with the performance requirements of 10 CFR 70.61 (CFR, 2003c).

The results of the ISA are presented in Section 3.6, Process Hazards; Section 3.7, General Types of Accident Sequences, and Section 3.8, IROFS.

B. LES has implemented programs to maintain the ISA and supporting documentation so that it is accurate and up-to-date. Changes to the ISA Summary are submitted to the NRC, in accordance with 10 CFR 70.72(d)(1) and (3) (CFR, 2003i). The ISA update process accounts for any changes made to the facility or its processes. This update will also verify that initiating event frequencies and IROFS reliability values assumed in the ISA remain valid. Any changes required to the ISA as a result of the update process will be included in a revision to the ISA. Evaluation of any facility changes or changes in the process safety information that may alter the parameters of an accident sequence is by the ISA method(s) as described in the ISA Summary Document. For any revisions to the ISA, personnel having qualifications similar to those of ISA team members who conducted the original ISA are used.

C. Personnel used to update and maintain the ISA and ISA Summary are trained in the ISA method(s) and are suitably qualified.

D. Proposed changes to the facility or its operations are evaluated by the ISA method(s) described in Section 3.1, General ISA Information. New or additional IROFS and appropriate management measures are designated as required. The adequacy of existing IROFS and associated management measures are promptly evaluated to determine if they are impacted by changes to the facility and/or its processes. If a proposed change results in a new type of accident sequence or increases the consequences or likelihood of a previously analyzed accident sequence within the context of 10 CFR 70.61 (CFR, 2003c), the adequacy of existing IROFS and associated management measures are promptly evaluated and the necessary changes are made, if required.

E. Unacceptable performance deficiencies associated with IROFS are addressed that are identified through updates to the ISA.

F. Written procedures are maintained on site.

G. All IROFS are maintained so that they are available and reliable when needed.

3.1.8.3 Management Measures

Management measures are functions applied to IROFS, and any items that may affect the function of IROFS. IROFS management measures ensure compliance with the performance requirements assumed in the ISA documentation. The measures are applied to particular structures, systems, equipment, components, and activities of personnel, and may be graded commensurate with the reduction of the risk attributable to that IROFS. The IROFS management measures shall ensure that these structures, systems, equipment, components, and activities of personnel within the identified IROFS boundary are designed, implemented, and maintained, as necessary, to ensure they are available and reliable to perform their function when needed, to comply with the performance requirements assumed in the ISA documentation.

The following types of management measures are required by the 10 CFR 70.4 definition of management measures. The description for each management measure reflects the general requirements applicable to each IROFS. Any management measure that deviates from the general requirements described in this section, which are consistent with the performance requirements assumed in the ISA documentation, are discussed in Section 3.8.3, Basis for Enhanced or High Availability Failure Probability Index Number. A cross reference from the associated IROFS in Table 3.8-1 to the applicable subsection is provided in Table 3.8-1.

<u>Configuration Management</u>

The configuration management program is required by 10 CFR 70.72 and establishes a system to evaluate, implement, and track each change to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel. Configuration management of IROFS, and any items that may affect the function of IROFS, is applied to all items identified within the scope of the IROFS boundary. Any change to structures, systems, equipment, components, and activities of personnel within the identified IROFS boundary must be evaluated before the change is implemented. If the change requires an amendment to the License, Nuclear Regulatory Commission approval is required prior to implementation.

<u>Maintenance</u>

Maintenance of IROFS, and any items that may affect the function of IROFS, encompasses planned surveillance testing and preventative maintenance, as well as unplanned corrective maintenance. Implementation of approved configuration management changes to hardware is also generally performed as a planned maintenance function.

Planned surveillance testing (e.g., functional/performance testing, instrument calibrations) monitors the integrity and capability of IROFS, and any items that may affect the function of IROFS, to ensure they are available and reliable to perform their function when needed, to comply with the performance requirements assumed in the ISA documentation. All necessary periodic surveillance testing is performed on an annual frequency (any exceptions credited within the ISA are discussed in Section 3.8.3).

Planned preventative maintenance (PM) includes periodic refurbishment, partial or complete overhaul, or replacement of IROFS, as necessary, to ensure the continued availability and reliability of the safety function assumed in the ISA documentation. In determining the frequency of any PM, consideration is given to appropriately balancing the objective of preventing failures through maintenance, against the objective of minimizing unavailability of IROFS because of PM. In addition, feedback from PM and corrective maintenance and the results of incident investigations and identified root causes are used, as appropriate, to modify the frequency or scope of PM.

Planned maintenance on IROFS, or any items that may affect the function of IROFS, that do not have redundant functions available, will provide for compensatory measures to be put into place to ensure that the IROFS function is performed until it is put back into service.

Corrective maintenance involves repair or replacement of equipment that has unexpectedly degraded or failed. Corrective maintenance restores the equipment to acceptable performance through a planned, systematic, controlled, and documented approach for the repair and replacement activities.

Following any maintenance on IROFS, and before returning an IROFS to operational status, functional testing of the IROFS, as necessary, is performed to ensure the IROFS is capable of performing its intended safety function.

Training and Qualifications

IROFS, and any items that may affect the function of IROFS, require that personnel involved at each level (from design through and including any assumed process implementation steps or actions) have and maintain the appropriate training and qualifications. Employees are provided with formal training to establish the knowledge foundation and on-the-job training to develop work performance skills. For process implemented steps or actions, a needs/job analysis is performed and tasks are identified to ensure that appropriate training is provided to personnel working on tasks related to IROFS. Minimum training requirements are developed for those positions whose activities are relied on for safety. Initial identification of job-specific training requirements is based on experience. Entry-level criteria (e.g., education, technical background, and/or experience) for these positions are contained in position descriptions.

Qualification is indicated by successful completion of prescribed training, demonstration of the ability to perform assigned tasks, and where required by regulation, maintaining a current and valid license or certification.

Continuing training is provided, as required, to maintain proficiency in specific knowledge and skill related activities. For all IROFS, and any items that may affect the function of IROFS, involving process implemented steps or actions, annual refresher training or requalification is required (any exceptions credited within the ISA are discussed in Section 3.8.3).

Procedures

All activities involving IROFS, and any items that may affect the function of IROFS, are conducted in accordance with approved procedures. Each of the other IROFS management measures (e.g., configuration management, maintenance, training) is implemented via approved procedures. These procedures are intended to provide a pre-planned method of conducting the activity in order to eliminate errors due to on-the-spot analysis and judgments.

All procedures are sufficiently detailed that qualified individuals can perform the required functions without direct supervision. However, written procedures cannot address all contingencies and operating conditions. Therefore, they contain a degree of flexibility appropriate to the activities being performed. Procedural guidance exists to identify the manner in which procedures are to be implemented. For example, routine procedural actions may not require the procedure to be present during implementation of the actions, while complex jobs, or checking with numerous sequences may require valve alignment checks, approved operator aids, or in-hand procedures that are referenced directly when the job is conducted.

To support the requirement to minimize challenges to IROFS, and any items that may affect the function of IROFS, specific procedures for abnormal events are also provided. These procedures are based on a sequence of observations and actions to prevent or mitigate the consequences of an abnormal situation.

Audits and Assessments

Audits are focused on verifying compliance with regulatory and procedural requirements and licensing commitments. Assessments are focused on effectiveness of activities and ensuring that IROFS are reliable and are available to perform their intended safety functions as documented in the ISA. The frequency of audits and assessments is based upon the status and safety importance of the activities being performed and upon work history. However, at a minimum, all activities associated with maintaining IROFS will be audited or assessed on an annual basis (any exceptions credited within the ISA are discussed in Section 3.8.3).

Incident Investigations

Incident investigations are conducted within the Corrective Action Program (CAP). Incidents associated with IROFS, and any items that may affect the function of IROFS, encompass a range of items, including (a) processes that behave in unexpected ways, (b) procedural activities not performed in accordance with the approved procedure, (c) discovered deficiency, degradation, or non-conformance with an IROFS, or any items that may affect the function of IROFS. Additionally, audit and assessment results are tracked in the Corrective Action Program.

Feedback from the results of incident investigations and identified root causes are used, as appropriate, to modify management measures to provided continued assurance that the reliability and availability of IROFS remain consistent with the performance requirements assumed in the ISA documentation.

Records Management

All records associated with IROFS, and any items that may affect the function of IROFS, shall be managed in a controlled and systematic manner in order to provide identifiable and retrievable documentation. Applicable design specifications, procurement documents, or other documents specify the QA records to be generated by, supplied to, or held, in accordance with approved procedures are included.

Other Quality Assurance Elements

Other quality assurance elements associated with IROFS, or any items that may affect the function of IROFS, that are required to ensure the IROFS is available and reliable to perform the function when needed to comply with the performance requirements assumed in the ISA documentation, will be listed in Table 3.8-1 and discussed in Section 3.8.3.

3.1.9 References

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NRC, 2001. Integrated Safety Analysis Guidance Document, NUREG-1513, U.S. Nuclear Regulatory Commission, May 2001.

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TABLES

Table 3.1-1 HAZOP Guidewords

UF_6 PROCESS GUIDEWORDS			
Less Heat	Corrosion	Maintenance	No Flow
More Heat	Loss of Services	Criticality	Reverse Flow
Less Pressure	Toxicity	Effluents/Waste	Less Uranium
More Pressure	Contamination	Internal Missile	More Uranium
Impact/Drop	Loss of Containment	Less Flow	Light Gas
Fire (Process, internal, other)	Radiation	More Flow	External Event
NON UF_6 PROCESS GUIDEWORDS			
High Flow	Low Pressure	Impact/Drop	More Uranium
Low Flow	High Temperature	Corrosion	External Event
No Flow	Low Temperature	Loss of Services	Startup
Reverse Flow	Fire	Toxicity	Shutdown
High Level	High Contamination	Radiation	Internal Missile
Low Level	Rupture	Maintenance	
High Pressure	Loss of Containment	Criticality	
No Flow			
EXTERNAL EVENTS POTENTIAL CAUSES			
Construction on Site	Hurricane	Seismic	Transport Hazard Off-Site
Flooding	Industrial Hazard Off-site	Tornado	External Fire
Airplane	Snow/Ice	Local Intense Precipitation	

Table 3.1-2 ISA HAZOP Table Sample Format
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ISA HAZOP NODE:		DESCRIPTION :			DATE:	PAGE:
GUIDEWORD	HAZARD	CAUSE	CONSEQUENCE	SAFEGUARDS	MITIGATING FACTORS	COMMENTS

Table 3.1-3 Consequence Severity Categories Based on 10 CFR 70.61
Page 1 of 1

	Workers	Offsite Public	Environment
Category 3 High Consequence	Radiation Dose (RD) >1 Sievert (Sv) (100 rem) For the worker (elsewhere in room), except the worker (local), Chemical Dose (CD) > AEGL-3 For worker (local), CD > AEGL-3 for HF CD > * for U	RD > 0.25 Sv (25 rem) 30 mg sol U intake CD > AEGL-2	–
Category 2 Intermediate Consequence	0.25 Sv (25 rem) <RD≤ 1 Sv (100 rem) For the worker (elsewhere in room), except the worker (local), AEGL-2 < CD≤ AEGL-3 For the worker (local), AEGL-2 < CD ≤ AEGL-3 for HF ** < CD ≤ * for U	0.05 Sv (5 rem) < RD≤ 0.25 Sv (25 rem) AEGL-1 <CD≤ AEGL-2	Radioactive release > 5000 x Table 2 Appendix B of 10 CFR Part 20
Category 1 Low Consequence	Accidents of lower radiological and chemical exposures than those above in this column	Accidents of lower radiological and chemical exposures than those above in this column	Radioactive releases with lower effects than those referenced above in this column

Notes:

*NUREG-1391 threshold value for intake of soluble U resulting in permanent renal failure

**NUREG-1391 threshold value for intake of soluble U resulting in no significant acute effects to an exposed individual

Table 3.1-4 Chemical Dose Information
Page 1 of 1

	High Consequence (Category 3)	**Intermediate Consequence (Category 2)**
Worker (local)	> 40 mg U intake > 139 mg HF/m^3	> 10 mg U intake > 78 mg HF/m^3
Worker (elsewhere in room)	> 146 mg U/m^3 > 139 mg HF/m^3	> 19 mg U/m^3 > 78 mg HF/m^3
Outside Controlled Area (30-min exposure)	> 13 mg U/m^3 > 28 mg HF/m^3	> 2.4 mg U/m^3 > 0.8 mg HF/m^3

Table 3.1-5 Likelihood Categories Based on 10 CFR 70.61
Page 1 of 1

	Likelihood Category	Probability of Occurrence*
Not Unlikely	3	More than 10^{-4} per-event per-year
Unlikely	2	Between 10^{-4} and 10^{-5} per-event per-year
Highly Unlikely	1	Less than 10^{-5} per-event per-year

*Based on approximate order-of-magnitude ranges

Table 3.1-6 Risk Matrix with Risk Index Values

Page 1 of 1

Severity of Consequences	Likelihood of Occurrence		
	Likelihood Category 1 Highly Unlikely (1)	Likelihood Category 2 Unlikely (2)	Likelihood Category 3 Not Unlikely (3)
Consequence Category 3 High (3)	Acceptable Risk 3	Unacceptable Risk 6	Unacceptable Risk 9
Consequence Category 2 Intermediate (2)	Acceptable Risk 2	Acceptable Risk 4	Unacceptable Risk 6
Consequence Category 1 Low (1)	Acceptable Risk 1	Acceptable Risk 2	Acceptable Risk 3

Table 3.1-7 (Not Used)

Table 3.1-8 Determination of Likelihood Category
Page 1 of 1

Likelihood Category	Likelihood Index T (= sum of index numbers)
1	$T \leq -5$
2	$-5 < T \leq -4$
3	$-4 < T$

Table 3.1-9 Failure Frequency Index Numbers

Frequency Index No.	Based On Evidence	Based On Type Of IROFS**	Comments
-6*	External event with freq. < 10^{-6} /yr		If initiating event, no IROFS needed.
-5*	Initiating event with freq. < 10^{-5}/yr		For passive safe-by-design components or systems, failure is considered highly unlikely when no potential failure mode (e.g., bulging, corrosion, or leakage) exists, as discussed in Section 3.1.1.3.2, significant margin exists*** and these components and systems have been placed under configuration management.
-4*	No failures in 30 years for hundreds of similar IROFS in industry	Exceptionally robust passive engineered IROFS (PEC), or an inherently safe process, or two independent active engineered IROFS (AECs), PECs, or enhanced admin. IROFS	Rarely can be justified by evidence. Further, most types of single IROFS have been observed to fail
-3*	No failures in 30 years for tens of similar IROFS in industry	A single IROFS with redundant parts, each a PEC or AEC	
-2*	No failure of this type in this facility in 30 years	A single PEC	
-1*	A few failures may occur during facility lifetime	A single AEC, an enhanced admin. IROFS, an admin. IROFS with large margin, or a redundant admin. IROFS	
0	Failures occur every 1 to 3 years	A single administrative IROFS	
1	Several occurrences per year	Frequent event, inadequate IROFS	Not for IROFS, just initiating events

Table 3.1-9 Failure Frequency Index Numbers
Page 2 of 2

Frequency Index No.	Based On Evidence	Based On Type Of IROFS**	Comments
2	Occurs every week or more often	Very frequent event, inadequate IROFS	Not for IROFS, just initiating events

*Indices less than (more negative than) –1 should not be assigned to IROFS unless the configuration management, auditing, and other management measures are of high quality, because, without these measures, the IROFS may be changed or not maintained.

**The index value assigned to an IROFS of a given type in column 3 may be one value higher or lower than the value given in column 1. Criteria justifying assignment of the lower (more negative) value should be given in the narrative describing ISA methods. Exceptions require individual justification.

***For components that are safe-by-volume, safe-by-diameter, or safe-by-slab thickness, significant margin is defined as a margin of at least 10%, during both normal and upset conditions, between the actual design parameter value of the component and the value of the critical design attribute. For components that require a more detailed criticality analysis, significant margin is defined as $k_{eff} < 0.95$, where $k_{eff} = k_{calc} + 3\sigma_{calc}$.

Table 3.1-10 Failure Probability Index Numbers

Probability Index No.	Probability of Failure on Demand	Based on Type of IROFS	Comments
-6*	10^{-6}		If initiating event, no IROFS needed.
-4 or -5*	10^{-4} - 10^{-5}	Exceptionally robust passive engineered IROFS (PEC), or an inherently safe process, or two redundant IROFS more robust than simple admin. IROFS (AEC, PEC, or enhanced admin.)	Can rarely be justified by evidence. Most types of single IROFS have been observed to fail
-3 or -4*	10^{-3} - 10^{-4}	A single passive engineered IROFS (PEC) or an active engineered IROFS (AEC) with high availability	
-2 or -3*	10^{-2} - 10^{-3}	A single active engineered IROFS, or an enhanced admin. IROFS, or an admin. IROFS for routine planned operations	
-1 or -2	10^{-1} - 10^{-2}	An admin. IROFS that must be performed in response to a rare unplanned demand	

*Indices less than (more negative than) –1 should not be assigned to IROFS unless the configuration management, auditing, and other management measures are of high quality, because, without these measures, the IROFS may be changed or not maintained.

Table 3.1-11 Failure Duration Index Numbers
Page 1 of 1

Duration Index No.	Avg. Failure Duration	Duration in Years	Comments
1	More than 3 yrs	10	
0	1 yr	1	
-1	1 mo	0.1	Formal monitoring to justify indices less than -1
-2	A few days	0.01	
-3	8 hrs	0.001	
-4	1 hr	10^{-4}	
-5	5 min	10^{-5}	

NATIONAL
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FACILITY

INTEGRATED SAFETY ANALYSIS SUMMARY

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3.2 SITE DESCRIPTION

This section provides an overall description of the National Enrichment Facility (NEF) site and its environment, including regional and local geography, demography, meteorology, hydrology, geology, seismology, and stability of subsurface materials. Significant portions of the information presented in this section were derived from the NEF Environmental Report (LES, 2003).

This section also provides a characterization of natural phenomena (e.g., tornadoes, hurricanes, floods, and earthquakes) and other external events (e.g., explosions and aircraft crashes) in sufficient detail to assess their impact on facility safety and to assess their likelihood of occurrence.

3.2.1 Site Geography

Site features are well suited for the location of an uranium enrichment facility as evidenced by favorable conditions of hydrology, geology, seismology and meteorology as well as good transportation routes for distributing feed and product by truck.

3.2.1.1 Site Location

The proposed NEF site is located in Southeastern New Mexico near the New Mexico/Texas state line, in Lea County. This location is about 8 km (5 mi) east of Eunice and about 32 km (20 mi) south of Hobbs. The site comprises about 220 ha (543 acres) and is within county Section 32, Township 21 South, Range 38 East. The approximate center of the NEF is at latitude 32 degrees, 26 min, 1.74 sec North and longitude 103 degrees, 4 min, 43.47 sec West (see Figure 3.2-1, County Map).

Section 32 is currently owned by the State of New Mexico. The State of New Mexico has granted a 35 year easement to LES for site access and control.

The NEF site is relatively flat with slight undulations in elevation ranging from 1,033 to 1,045 m (3,390 to 3,430 ft) above mean sea level. The overall slope direction is to the southwest. Except for a gravel covered road which bisects the east and west halves of Section 32, the property is undeveloped and utilized for domestic livestock grazing (see Figure 3.2-2, Plot Plan).

Figure 3.2-3, Site Plan, shows the site property boundary and the general layout of the buildings.

3.2.1.2 Public Roads and Transportation

3.2.1.2.1 Public Roads

The site lies along the north side of New Mexico Highway 234. New Mexico Highway 234 intersects New Mexico Highway 18 about 4 km (2.5 mi) to the west. (See Figure 3.2-1). To the north, U.S. Highway 62/180 intersects New Mexico Highway 18 providing access from the city of Hobbs south to New Mexico Highway 234. To the east in Texas, U.S. Highway 385 intersects Texas Highway 176 providing access from the town of Andrews west to New Mexico Highway 234. To the south in Texas, Interstate 20 intersects Texas Highway 18 which becomes New Mexico Highway 18. West of the site, New Mexico Highway 8 provides access from the city of Eunice east to New Mexico Highway 234.

Potential adverse impact to NEF from chemical releases or explosions from trucks on nearby highways was evaluated. Due to the distance of the highway from the facility boundary, a chemical release from a passing vehicle will not have a safety impact on facility operations. Detailed probabilistic analyses show the annual probability of an explosion adversely impacting the plant is less than 1.0 E-5 per year.

3.2.1.2.2 Railroads

The nearest active rail transportation (the Texas-New Mexico Railroad) is in Eunice, New Mexico to the west about 5.8 km (3.6 mi) from the site. This rail line is used mainly by the local oil and gas industry for freight transport. There is also a rail spur to the Waste Control Specialists (WCS) facility along the northern boundary of the NEF site about 1 km (0.5 mi) from the Separations Building. This spur does not transport explosive materials or chemical shipments which could have a safety impact on facility operations. As such, there is no railroad traffic within proximity to the facility which poses a safety concern.

3.2.1.2.3 Water Transportation

There are no navigable waterways in the vicinity of the site.

3.2.1.2.4 Air Transportation

The nearest airport facilities are located just west of Eunice and are maintained by Lea County. The airport is about 16 km (10 mi) west of the proposed NEF and consists of two runways measuring about 1,000 m (3,280 ft) and 780 m (2,550 ft) each. Privately owned planes are the primary users of the airport. There is no control tower and no commercial air carrier flights (DOT, 2003). The nearest major commercial carrier airport is Lea County Regional Airport in Hobbs, New Mexico, about 32 km (20 mi) north.

An aircraft hazard analysis has been performed for the facility site, following the methodology of NUREG-0800 (NRC, 1981). Airports and airways in the vicinity of the site have been identified. Based on the published number of operations and distance to the proposed site, it is concluded that the presence of these airports does not pose any risk to the site with regard to aircraft hazard. For the identified airways, the probability of aircraft along these airways crashing onto the proposed site has been conservatively calculated to be less than 1.0 E-6 per year.

3.2.1.3 Nearby Bodies of Water

The climate in southeast New Mexico is semi-arid. Average precipitation at the site is calculated to be 33 to 38 cm (13 to 15 in) per year. Evaporation and transpiration rates are high. This results in minimal, if any, surface water occurrence.

The NEF site contains no surface drainage features. The site topography is relatively flat. Some localized depressions exist due to eolian processes, but the size of these features is too small to be of significance with respect to surface water collection.

The closest water conveyance is Monument Draw, a typically dry, intermittent stream located several miles west of the site.

Baker Spring, an intermittent surface water feature, is situated a little over 1.6 km (1 mi) northeast of the NEF site.

There are also three "produced water" lagoons for industrial purposes on the adjacent quarry property to the north.

There is also a manmade pond at the Eunice golf course approximately 15 km (9.5 mi) west of the site.

3.2.2 Demographics and Land Use

This section provides the census results for the site area, specific information about nearby population areas with respect to proximity to the site, specific information about nearby public facilities (schools, hospitals, parks, etc.) with respect to proximity to the site, and land and water use near the site.

3.2.2.1 Population Information

This section describes the population characteristics of the two-county areas around the NEF site.

3.2.2.1.1 Permanent Population and Distribution

The combined population of the two counties in the NEF vicinity, based on the 2000 U.S. Census is 68,515, which represents a 2.3% decrease over the 1990 population of 70,130 (Table 3.2-1, Population and Population Projections, 1970-2040). This rate of decrease is counter to the trends for the states of New Mexico and Texas, which had population increases of 20.1% and 22.8%, respectively during the same decade. Over that 10 year period, Lea County, New Mexico, where the site is located, had a growth decrease of 0.5% and the Andrews County, Texas decrease was 9.3%. Lea County experienced a sharp but short population increase in the mid-1980's due to petroleum industry jobs. The change in the job market caused the population in Lea County to increase to over 65,000 during that period.

Based on projections made using historic data (Table 3.2-1), Lea County, New Mexico and Andrews County, Texas are likely to grow more slowly than their respective states over the next 30 years (the expected licensed period for the NEF).

Lea County covers 11,378 km^2 (4,393 mi^2) or approximately 1,142,238 ha (2,822,522 acres) which is three times the size of Rhode Island and only slightly smaller than Connecticut. The county population density is 16% lower than the New Mexico state average (4.8 versus 5.8 people per square kilometer (12.6 versus 15.0 people per square mile)). The county housing density is 20% lower than the New Mexico state average (2.0 versus 2.5 housing units per square kilometer (5.3 versus 6.4 housing units per square mile)).

Andrews County covers 3,895 km^2 (1,504 mi^2). The county population density is 11% of the Texas state average (3.3 versus 30.6 per square kilometer (8.7 versus 79.6 population density per square mile)). The county housing density is low, at just over 11% of the Texas state average (1.4 versus 12.0 housing units per square kilometer (3.6 versus 31.2 housing units per square mile)).

3.2.2.1.2 Industrial Population

More than 98% of the area within an 8 km (5 mi) radius of the NEF is an extensive area of open land on which livestock wander and graze. Gas and oil field operations are widespread in the area, but significant petroleum potential is absent within at least 5 to 8 km (3 to 5 mi) of the site. Industrial operations near the site include:

- A quarry, operated by Wallach Concrete, Inc., and several oil recovery sludge ponds owned by the Sundance Services are located north of the site. The quarry owner leases land space to a "produced water" reclamation company that maintains three small "produced water" lagoons. Eight people are employed at the Wallach Concrete Quarry and nine people are employed by Sundance Services.
- Lea County operates a landfill on the south side of New Mexico State Highway 234, approximately 1 km (0.6 mi) from the center of Section 32. Four people are employed at the Lea County landfill.
- A vacant parcel of land is immediately east of the site. Land further east approximately 1.6 km (1 mi), in Texas, is occupied by Waste Control Specialists (WCS), LLC. WCS possesses a radioactive materials license from Texas, an NRC Agreement state. WCS is licensed to treat and temporarily store low-level and mixed low-level radioactive waste. WCS is also permitted to treat and dispose of hazardous toxic waste in a landfill. WCS employs 72 people.
- Dynegy's Midstream Services Plant is located 6 km (4 mi) from the site. This facility is engaged in the gathering and processing of natural gas. The Dynegy Midstream Services Plant employs 40 people.

3.2.2.2 Population Centers

The proposed NEF site is in Lea County, New Mexico, approximately 1 km (0.6 mi) from the border of Andrews County, Texas, as shown on Figure 3.2-1. The figure also shows the city of Eunice, New Mexico, the closest population center to the site, at a distance of about 8 km (5 mi). Other population centers are at distances from the site as follows:

- Hobbs, Lea County, New Mexico: 32 km (20 mi) north
- Jal, Lea County, New Mexico: 37 km (23 mi) south
- Lovington, Lea County New Mexico: 64 km (39 mi) north-northwest

- Andrews, Andrews County Teas: 51 km (32 mi) east
- Seminole, Gaines County Texas, 51 km (32 mi) east-northeast
- Denver City, Gaines County, Texas 65 km (40 mi) north-northeast.

Aside from these communities, the population density in the site region is extremely low. Table 3.2-1, lists by year/decade, the estimated population in the site vicinity.

3.2.2.3 Public Service Facilities

3.2.2.3.1 Fire Department and Local Law Enforcement

Fire support service for the Eunice area is provided by Eunice Fire and Rescue, located approximately 8 km (5 mi) from the site. It is staffed by one full-time fire chief and 34 volunteer firefighters. Fire fighting equipment includes three pumpers, one tanker and three grass trucks. If additional fire equipment is needed, or if Eunice Fire and Rescue is unavailable, mutual aid agreements exist with all of the county fire departments.

The Eunice Police Department, with five full-time officers, provides local law enforcement. The Lea County Sheriff's Department also maintains a substation in Eunice. If additional resources are needed, officers from mutual aid communities within Lea County and Andrews County, Texas, can provide an additional level of response. The New Mexico State Police provide a third level of response.

3.2.2.3.2 School Population

There are four educational institutions within a radius of about 8 km (5 mi) of the NEF site, all in Lea County, New Mexico. These include an elementary school, a middle school, a high school and a private K-12 school. Table 3.2-2, Educational Facilities Near the Site, details the location of the educational facilities, population (including faculty/staff members), and student-teacher ratio. Apart from these schools, the next closest educational institutions are in Hobbs, New Mexico, 32 km (20 mi) north of the site.

The closest schools in Andrews County, Texas are in the community of Andrews about 51 km (32 mi) east of the NEF site.

3.2.2.3.3 Health Care Populations

There are two hospitals in Lea County, New Mexico. The Lea Regional Medical Center is located in Hobbs, New Mexico, about 32 km (20 mi) north of the proposed NEF site. This 250-bed hospital can handle acute and stable chronic care patients. In Lovington, New Mexico, 64 km (39 mi) north-northwest of the site, Covenant Medical Systems manages Nor-Lea Hospital, a full-service, 27-bed facility.

There are no nursing homes or retirement facilities in the site area. The closest such facilities are in Hobbs, New Mexico, about 32 km (20 mi) north of the site.

3.2.2.3.4 Recreational Population

There are no recreational facilities near the site. The Eunice Golf Course is located approximately 15 km (9.2 mi) from the site. A historical marker and picnic area is located about 3.2 km (2 mi) from the site at the intersection of New Mexico Highways 234 and 18.

3.2.2.4 Industrial Areas

More than 98% of the area within an 8 km (5 mi) radius of the NEF is an extensive area of open land on which livestock wander and graze. Gas and oil field operations are widespread in the area, but significant petroleum potential is absent within at least 5 to 8 km (3 to 5 mi) of the site. Industrial operations near the site include:

- A quarry, operated by Wallach Concrete, Inc., and several oil recovery sludge ponds owned by the Sundance Services are located north of the site. The quarry owner leases land space to a "produced water" reclamation company that maintains three small "produced water" lagoons. The operations at these facilities do not pose a safety concern for the NEF.
- Lea County operates a landfill on the south side of New Mexico State Highway 234, approximately 1 km (0.6 mi) from the center of Section 32. This facility does not pose a safety concern for the NEF.
- A vacant parcel of land is immediately east of the site. Land further east approximately 1.6 km (1 mi), in Texas, is occupied by WCS. WCS possesses a radioactive materials license from Texas, an NRC Agreement state. WCS is licensed to treat and temporarily store low-level and mixed low-level radioactive waste. WCS is also permitted to treat and dispose of hazardous toxic waste in a landfill. WCS does not pose a safety concern for the NEF.
- Dynegy's Midstream Services Plant is located 6 km (4 mi) from the site. This facility is engaged in the gathering and processing of natural gas.
- An underground CO2 pipeline currently traverses the property in a southeast-northwest direction. The 254 mm (10 in) diameter pipe operates at 134.4 bar (1,950 psi). The pipeline will be relocated along the western and southern boundary of Section 32 so that it will be at least 396.2 m (1,300 ft) from the facility Restricted Area. At this distance from the facility, the pipeline does not pose a safety concern.
- An underground natural gas pipeline is located along the south property line, paralleling New Mexico Highway 234. A risk assessment of the hazards posed by the pipeline has been performed. The assessment used a hazard model to estimate the likelihood of a gas line leak and subsequent explosion that could impact NEF operations. The model incorporated historical data on pipeline accidents obtained from the Department of Transportation (DOT, 2002) and accounted for the conditional probability that if an explosion were to occur, it would have to be substantial to have an impact on facility buildings. The model also accounted for the safe separation distance, i.e., if an explosion occurs beyond the safe separation distance for a critical structure, then the structure will be unaffected. The calculated probability of the hazard due to the natural gas pipeline in the vicinity of the proposed NEF is 9.4 E-6 per year.

3.2.2.5 Land Use

Surrounding property consists of vacant land and industrial developments. A railroad spur borders the site to the north. Beyond is a sand/aggregate quarry. A vacant parcel of land is situated immediately to the east. Cattle grazing are not allowed on this vacant parcel. Further east, at the state line and within Andrews County, Texas, is a hazardous waste treatment and disposal facility. A landfill is south-southeast of the site, across New Mexico Highway 234 and a petroleum contaminated soil treatment facility is adjacent to the west. Land further north, south and west has been mostly developed by the oil and gas industry. Land further east is ranchland. The nearest residences are situated approximately 4.3 km (2.63 mi) west of the site. Beyond is the city of Eunice, which is approximately 8 km (5 mi) to the west. There are no known public recreational areas with 8 km (5 mi) of the site. There is a historical marker and picnic area approximately 3.2 km (2 mi) from the site at the intersection of New Mexico Highways 234 and 18. Refer to Section 3.2.5.2 for further discussion on mineral resources in the site vicinity.

Rangeland comprises 98.5% of the area within an 8 km (5 mi) radius of the NEF site, encompassing 12,714 ha (31,415 acres) within Lea County, New Mexico, and 7,213 ha (17,823 acres) in Andrews County, Texas. Rangeland is an extensive area of open land on which livestock wander and graze and includes herbaceous rangeland, shrub and brush rangeland and mixed rangeland. Built-up land and barren land constitute the other two land use classifications in the site vicinity, but at considerably smaller percentages. Land cover due to built-up areas, which includes residential and industrial developments, makes up 1.2 percent of the land use. This equates to a combined total of 243 ha (601 acres) for Lea and Andrews Counties. The remaining 0.3% of land area is considered barren land which consists of bare exposed rock, transitional areas and sandy areas. This information is summarized in Table 3.2-3, Land Use Within 8 km (5 mi) of the Site. The above indicated land use classifications are identical to those used by the United States Geological Survey (USGS). No special land use classifications (i.e., Native American reservations, national parks, prime farmland) are within the vicinity of the site.

Except for the proposed construction of the NEF and the potential citing of a low-level radioactive waste disposal site in Andrews County, Texas, there are not other know current, future or proposed land use plans, including staged plans, for the site or immediate vicinity.

3.2.2.6 Water Use

The climate in southeast New Mexico is semi-arid. Average precipitation at the site is calculated to be only 33 to 38 cm (13 to 15 in) per year. The NEF site itself contains no surface water bodies or surface drainage features. Essentially all the precipitation that occurs at the site is subject to infiltration and/or evapotranspiration.

3.2.2.6.1 Recreation

There are no significant bodies of water or navigable waterways in the vicinity of the site.

3.2.2.6.2 Agricultural Water Use

Although various crops are grown within Lea and Andrews Counties, local and county officials report that there is no agricultural activity in the site vicinity, except for domestic livestock ranching. The principal livestock for both Lea and Andrews Counties is cattle. Although milk cows comprise a significant number of cattle in Lea County, the nearest dairy farms are about 32 km (20 mi) north of the subject site, near the city of Hobbs, New Mexico. There are no milk cows in Andrews County. Table 3.2-4, Agriculture Census, Crop, and Livestock Information, provides data on agricultural and livestock activities in Lea County, New Mexico, and Andrews County, Texas.

Known sources of water in the site vicinity include the following: a manmade pond on the adjacent quarry property to the north which is stocked with fish for private use; Baker Spring, an intermittent surface water feature, situated a little over 1.6 km (1 mi) northeast of the site which only contains water seasonally; several cattle watering holes where groundwater is pumped by windmill and stored in above ground tanks.

3.2.2.6.3 Municipal Use of Local Surface Water

Surface water is not a source of water for municipal use.

3.2.2.6.4 Groundwater Use

The NEF water supply is from the municipal water systems in Hobbs and Eunice, New Mexico, and thus no water will be drawn from either surface water or groundwater sources at the NEF site. The Eunice system obtains water from a groundwater source in the city of Hobbs, approximately 32 km (20 mi) north of the site. Supply of nearby groundwater users will thus not be affected by operation of the NEF. No subsurface or surface water uses such as withdrawals or consumption are made at the site by the NEF.

3.2.3 Meteorology

In this section, data characterizing the meteorology (e.g., winds, precipitation, and severe weather) for the site are presented. The discussion identifies the design basis natural events for the facility, including the likelihood of occurrence.

The meteorological conditions at the NEF have been evaluated and summarized in order to characterize the site climatology and to provide a basis for predicting the dispersion of gaseous effluents. No on-site meteorological data were available, however, WCS have a meteorological monitoring station within approximately 1.6 km (1 mi) from the proposed NEF site.

Climate information from Hobbs, New Mexico (32 km (20 mi) north of the site), obtained from the Western Regional Climate Center, were used. In addition, National Oceanic and Atmospheric Administration (NOAA) Local Climatological Data (LCD) recorded at Midland-Odessa Regional Airport, Texas (103 km (64 mi) southeast of the site) and at Roswell, New Mexico (161 km (100 mi) northwest of the site) were used. In the following summaries of meteorological data, the averages are based on:

- Hobbs station (WRCC, 2003) averages are based on a 30 year record (1971 to 2000) unless otherwise stated

- Midland-Odessa station (NOAA, 2002a) averages are based on a 30 year record (1961 to 1990) unless otherwise stated
- Roswell station (NOAA, 2002b) averages are based on a 30 year record (1961 to 1990) unless otherwise stated.

The WCS data was not used since it had not been fully verified by WCS. An analysis of the WCS data was performed and it was determined that the prevailing wind direction at the WCS facility agrees with the prevailing wind directions at Midland-Odessa and Roswell. Use of the Hobbs, Midland-Odessa, and Roswell observations for a general description of the meteorological conditions at the NEF was deemed appropriate as they are all located within the same region and have similar climates. Use of the Midland-Odessa data for predicting the dispersion of gaseous effluents was deemed appropriate. It is the closest first-order National Weather Service (NWS) station to the NEF site, and both Midland-Odessa and the NEF site have similar climates. In addition, wind direction frequency comparisons between Midland-Odessa and the closest source of meteorological measurements (WCS) to the NEF site show good agreement. Midland-Odessa and Roswell data were compiled and certified by the National Climatic Data Center. Hobbs data were compiled and certified by the Western Regional Climate Center.

3.2.3.1 Local Wind Patterns and Average and Maximum Wind Speeds

Monthly mean wind speeds and prevailing wind directions at Midland-Odessa are presented in Table 3.2-5, Midland-Odessa, Texas, Wind Data. The annual mean wind speed was 4.9 m/s (11.0 mi/hr) and the prevailing wind direction was 180 degrees with respect to true north. The maximum five-second wind speed was 31.3 m/s (70 mi/hr).

Monthly mean wind speeds and prevailing wind directions at Roswell are presented in Table 3.2-6, Roswell, New Mexico, Wind Data. The annual mean wind speed was 3.7 m/s (8.2 mi/hr) and the prevailing wind direction was wind from 160 degrees with respect to true north. The maximum five-second wind speed was 27.7 m/s (62 mi/hr).

Five years of data (1987-1991) from the Midland-Odessa NWS were used to generate joint frequency distributions of wind speed and direction. This data summary, for all Pasquill stability classes (A-F) combined, is provided in Table 3.2-7, Midland-Odessa Five Year (1987-1991) Annual Joint Frequency Distribution For All Stability Classes Combined.

Five years of data (1987-1991) from the Midland-Odessa NWS were used to generate joint frequency distributions of wind speed and direction as a function of Pasquill stability class (A-F). Stability class was determined using the solar radiation/cloud cover method. These data are given in Tables 3.2-8 through 3.2-13. The most stable classes, E and F, occur 18.3% and 13.6% of the time, respectively. The least stable class, A, occurs 0.4% of the time. Important conditions for atmospheric dispersion, stable (Pasquill class F) and low wind speeds 0.4-1.3 m/s (1.0-3.0 mi/hr), occur 2.2% of the time. The highest occurrences of Pasquill class F and low wind speeds, 0.4-1.3 m/s (1.0-3.0 mi/hr), with respect to wind direction are 0.28% and 0.23% with south and south-southeast winds.

3.2.3.2 Annual Amounts and Forms of Precipitation

The normal annual total rainfall as measured in Hobbs is 46.1 cm (18.15 in). Precipitation amounts range from an average of 1.2 cm (0.45 in) in March to 8 cm (3.1 in) in September. The record maximum and minimum monthly totals are 35.13 cm (13.83 in) and zero, respectively (WRCC, 2003). Table 3.2-14, Hobbs New Mexico Temperature and Precipitation Data, lists the monthly averages and extremes of precipitation for the Hobbs data. These precipitation summaries are based on 30 year records.

The normal annual total rainfall as measured in Midland-Odessa is 37.6 cm (14.8 in). Precipitation amounts range from an average of 1.1 cm (0.42 in) in March to 5.9 cm (2.31 in) in September. The record maximum and minimum monthly totals are 24.6 cm (9.70 in) and zero, respectively. The highest 24-hour precipitation total was 15.2 cm (6 in) in July 1968 (NOAA, 2002a). Table 3.2-15, Midland-Odessa, Texas, Precipitation Data, lists the monthly averages and extremes of precipitation for the Midland-Odessa data. These precipitation summaries are based on 30 year records.

The normal annual rainfall total as measured in Roswell, New Mexico, is 33.9 cm (13.34 in). The record maximum and minimum monthly totals are 17.5 cm (6.9 in) and zero, respectively (NOAA, 2002b, 2002a). The highest 24-hour precipitation total was 12.5 cm (4.91 in) in July 1981 (NOAA, 2002b). Table 3.2-16, Roswell, New Mexico, Precipitation Data, lists the monthly averages and extremes of precipitation for the Roswell data. These precipitation summaries are based on 30 year records.

3.2.3.3 Design Basis Values for Snow or Ice Load

Snowfall in Midland-Odessa, Texas, averages 13.0 cm (5.1 in) per year. Maximum monthly snowfall/ice pellets of 24.9 cm (9.8 in) fell in December 1998. The maximum amount of snowfall/ice pellets to fall in 24 hours was 24.9 cm (9.8 in) in December 1998 (NOAA, 2002a). Table 3.2-17, Midland-Odessa, Texas, Snowfall Data, lists the monthly averages and maximums of snowfall/ice pellets at Midland-Odessa, Texas. These snowfall summaries are based on 30 year records.

Snowfall in Roswell, New Mexico, averages 30.2 cm (11.9 in) per year. Maximum monthly snowfall/ice pellets of 53.3 cm (21.0 in) fell in December 1997. The maximum amount of snowfall/ice pellets to fall in 24 hours was 41.9 cm (16.5 in) in February 1988 (NOAA, 2002b). Table 3.2-18, Roswell, New Mexico, Snowfall Data, lists the monthly averages and maximums of snowfall/ice pellets at Roswell, New Mexico. These snowfall summaries are based on 30 year records.

The design basis snow load for the NEF was determined by combining the 100-year snowpack loading and 48 hour Probable Maximum Winter Precipitation (PMWP) loading for the area. Using the published 50 year snowpack loading of 48.8 kg/m^2 (10 lb/ft^2) (ASCE, 1998) and adjusting this value using the method described by ASCE, the 100 year snowpack loading is determined to be 58.6 kg/m^2 (12 lb/ft^2).

The 48-hour PMWP as determined by the methodology outlined in Hydrometeorlogical Report No. 33 (WB, 1956) is determined to be 483 mm (19 in), which corresponds to a loading of 96.6 kg/m^2 (19.8 lb/ft^2). These two values were used to develop a design basis snow loading of 156 kg/m^2 (32 lb/ft^2).

The design basis snow load does not explicitly account for loads due to frozen rain, ice, or hail. This type of loading is bounded by the conservative design basis snow load discussed above.

3.2.3.4 Type, Frequency, and Magnitude of Severe Weather

This section identifies the design basis severe weather events for the facility and describes the basis for their selection.

3.2.3.4.1 Tornados and Tornado Missiles

Tornadoes occur infrequently in the vicinity of the NEF. Only two significant tornadoes (i.e., F2 or greater) were reported in Lea County, New Mexico, (Grazulis, 1993) from 1880-1989. Across the state line, only one significant tornado was reported in Andrews County, Texas, (Grazulis, 1993) from 1880-1989.

Tornadoes are commonly classified by their intensities. The F-Scale classification of tornados is based on the appearance of the damage that the tornado causes. There are six classifications, F0 to F5, with an F0 tornado having winds of 64-116 km/hr (40-72 mi/hr) and an F5 tornado having winds of 420–512 km/hr (261-318 mi/hr) (AMS, 1996). The two tornadoes reported in Lea County were estimated to be F2 tornadoes (Grazulis, 1993).

The following steps were taken in performing the tornado hazard assessment for the site:

- Define a local region of latitude and longitude that surrounds the site of interest and obtain historical records of tornadoes that have touched down in the local region
- Determine occurrence rate and associated confidence limits
- Determine number of tornadoes per F-Scale category
- Estimate the damage path area for each F-Scale category and calculate damage areas associated with confidence limits
- Calculate tornado hazard probabilities for each F-Scale wind speed category.

An annual tornado hazard probability of 1E-05 was chosen for the design basis tornado. The tornado and tornado missile parameters from the site-specific study are provided below.

Annual Tornado Hazard Probability	1E-05
Tornado Wind Speed	302 km/hr (188 mi/hr)
Radius of Damaging Winds	130 m (425 ft)
Atmospheric Pressure Change (APC)	390 kg/m^2(80 lb/ft^2)
Rate of APC	146 kg/m^2/s (30 lb/ ft^2)

Missile: 2x4 Timber Plank, 6.80 kg (15 lb)

Horizontal Speed	136 km/hr (85 mi/hr)
Vertical Speed	88 km/hr (55 mi/hr)

Maximum Height above Ground	61 m (200 ft)
Missile: 76.2 mm (3-in.) Diameter Steel Pipe, 34 kg (75 lb)	
Horizontal Speed	80 km/hr (50 mi/hr)
Vertical Speed	48 km/hr (30 mi/hr)
Maximum height above Ground	9.1 m (30 ft)
Missile: Automobile 1361 kg (3,000 lb)	
Horizontal Speed	32 km/hr (20 mi/hr)

3.2.3.4.2 Extreme Winds

Annual extreme winds recorded at the Midland-Odessa, Texas, airport are used to model the straight wind hazard at the NEF site. The airport is located 103 km (64 mi) east-southeast of the site. The airport location features flat, open terrain. Due to proximity, common weather systems affect Eunice, New Mexico, and Midland-Odessa, Texas. The wind speeds used in the model are 3 second gust speeds at a 10 m height above ground. The set of annual extreme winds include the years 1973 to 1999.

A Fischer-Tippett Type I extreme value distribution is fit to the annual extreme wind speed data. Upper and lower bound values at 95% confidence level are also calculated. The results of the straight wind hazard assessment are provided in Table 3.2-19, Straight Wind Hazard Assessment.

An annual wind hazard probability of 1E-05 was chosen for the design basis wind speed. This wind speed is 252 km/hr (157 mi/hr), and is a 3 second gust, 10 m (33 ft) above ground.

3.2.3.4.3 Hurricanes

Hurricanes, or tropical cyclones, are low-pressure weather systems that develop over the tropical oceans. These storms are classified during their life cycle according to their intensity:

- Tropical depression – wind speeds less than 63 km/hr (39 mi/hr)
- Tropical storm – wind speed between 63 and 118 km/hr (39 and 73 mi/hr)
- Hurricane – wind speeds greater than 118 km/hr (73 mi/hr)

Hurricanes are fueled by the relatively warm tropical ocean water and lose their intensity quickly once they make landfall. Since the NEF is sited about 805 km (500 mi) from the coast, it is most likely that any hurricane that is tracked towards it would have dissipated to the tropical depression stage, that is, wind speeds less than 63 km/hr (39 mi/hr), before it reached the NEF. Therefore hurricanes are not a design basis event for the site.

3.2.3.4.4 Extreme Precipitation

The short duration – small area local intense probable maximum precipitation (PMP) was obtained from NOAA Hydrometeorological Report No. 52 (NOAA, 1982). The local intense PMP is 43.9 cm (17.3 in) in 1 hr over 2.6 km^2 (1 mi^2).

Roofs will be designed so as not to pond water to a depth during the local intense PMP that could exceed the design load for the roof. This will be accomplished by designing the parapets to a height which will preclude significant ponding on the roof. As an alternative, the parapets can be provided with scuppers that are designed to preclude significant roof ponding during the local intense PMP.

Local site runoff will be determined for the local plant site drainage area. Maximum ponding depths around the main plant structures will be determined using final site topography. The potential for water intrusion into critical plant areas will be precluded by final site grading.

3.2.3.4.5 Lightning

Thunderstorms occur during every month but are most common in the spring and summer months. Thunderstorms occur an average of 36.4 days/year in Midland-Odessa, Texas, based on a 54 year period of record. The seasonal averages are: 11 days in spring (March through May); 17.4 days in summer (June through August); 6.7 days in fall (September through November); and 1.3 days in winter (December through February).

J. L. Marshall (Marshall, 1973) presented a methodology for estimating lightning strike frequencies which includes consideration of the attractive area of structures. His method consists of determining the number of lightning flashes to earth per year per square kilometer and then defining an area over which the structure can be expected to attract a lightning strike. Assuming that there are 4 flashes to earth per year per square kilometer (2.1 flashes to earth per year per square mile) in the vicinity of the NEF (conservatively estimated using Figure 3.2-4, Average Lightning Flash Density, which is taken from the NWS (NWS, 2003). Marshall defines the total attractive area, A, of a structure with length L, width W, and height H, for lightning flashes with a current magnitude of 50% of all lightning flashes as:

$$A = LW + 4H (L + W) + 12.57 H^2$$

The following building complex dimensions were used to estimate conservatively the attractive area of the NEF:

L = 534 m (1,752 ft), W = 534 m (1,752 ft), H = 13 m (43 ft)

The total attractive area is therefore equal to 0.34 km^2 (0.13 mi^2). Consequently, the lightning strike frequency computed using Marshall's methodology is given as 1.36 flashes per year.

Lightning protection for the NEF is provided.

3.2.4 Hydrology

This section describes the NEF site's surface water and groundwater resources. Data is provided for the NEF site and the surrounding area, and the regional associations of those natural water systems are described. This information provides the basis for evaluation of any potential facility impacts on surface water, aquifers, and the related social and economic structures of the area around the facility.

The information included in this section was largely obtained from prior site studies including extensive subsurface investigations for a nearby facility, WCS, located about 1.6 km (1 mi) to the east of the NEF site. In addition, literature searches were conducted to obtain additional reference material. Some of the WCS data has been collected on Section 33 located immediately east of the NEF site. These data are being supplemented by a groundwater exploration and sampling program on Section 32 initiated by LES in September 2003.

The NEF facility will make no use of either surface water or groundwater from the site. The collection and storage of runoff from specific site areas will be controlled. No significant adverse changes are expected in site hydrology as a result of construction or operation of the NEF.

3.2.4.1 Surface Hydrology

The NEF site itself contains no surface water bodies or surface drainage features. Essentially all the precipitation that occurs at the site is subject to infiltration and/or evapotranspiration. More information on the movement and fate of surface water and groundwater at the site is provided in the following sections.

3.2.4.2 Major Surface and Subsurface Hydrological Systems

The climate in southeast New Mexico is semi-arid. Average precipitation at the site is calculated to be 33 to 38 cm per year (13 to 15 in per year). Evaporation and transpiration rates are high. This results in minimal, if any, surface water occurrence or groundwater recharge.

The NEF site is relatively flat and contains no surface drainage features.. Some localized depressions exist, due to eolian processes, but the size of these features is too small to be of significance with respect to surface water collection.

Most precipitation is contained onsite due to infiltration and/or evapotranspiration. The vegetation on the site is primarily mesquite bush (Prosopis juliflora) and native grasses (e.g., Sporobolus giganteus). The surface soils are predominantly of an alluvial or eolian origin. The texture of the surface soils is generally silt to silty sands. Therefore, the surface soils are relatively low in permeability and tend to hold moisture in storage rather than allow rapid infiltration to depth. Water held in storage in the soil is subsequently subject to evapotranspiration. Nine subsurface borings were drilled at the site during September 2003. Only one of the borings produced cuttings that were slightly moist at 1.8 to 4.2 m (6 to 14 ft) below ground surface; other cuttings were very dry. Evapotranspiration processes are significant enough to short-circuit any potential groundwater recharge. This process is further discussed below.

There is some evidence for shallow, near-surface groundwater occurrence in areas to the north and east of the site. These conditions are intermittent and limited. A quarry operated by Wallach Concrete, Inc. is located just north of the NEF site. Wallach Concrete has extensively mined sand and gravel from the quarry. The typical geologic cross section at that site consists of a layer of caliche at the surface, referred to as the "caprock," underlain by a sand and gravel deposit, which in turn overlies a thick clay unit of the Dockum Group, referred to as red beds, and part of the Chinle Formation. Figure 3.2-5, Site Boring Plan and Profile, depicts this stratigraphy. In some locations, the caprock (caliche) overlies sand and gravel, with the red bed clay Chinle Formation at the base of the pit. In some areas the caprock is missing and the sand and gravel is exposed at the surface. The caprock is generally fractured and following precipitation events may allow infiltration that quickly bypasses any roots from surface vegetation. In addition, gravel outcrops may allow rapid infiltration of precipitation. These conditions have led to instances of minor amounts of perched groundwater at the base of the sand and gravel unit, atop the red bed Chinle Formation. The Chinle red bed clay has a very low permeability, about 1×10^{-8} cm/s (4×10^{-9} in/s) (Rainwater, 1996), and serves as a confining unit arresting downward percolation of localized recharge flux. This shallow perched zone is not pervasive throughout the area.

Conditions at the NEF site are different than at the Wallach Concrete site. Two differences are of particular importance. First, the caprock is not present at the NEF site. Therefore, rapid infiltration through fractured caliche does not contribute to localized recharge at the NEF site. Second, the surface soils at the NEF site are finer-grained than the sand and gravel at the Wallach Concrete site. There is a thin layer of sand and gravel just above the red bed Chinle clay unit on the NEF site, but based on recent investigations, it is not saturated.

Another instance of possible saturation above the Chinle clay may be seen at Baker Spring, just to the northeast of the NEF site. Baker Spring is located at the edge of an escarpment, where the caprock ends. Baker Spring is intermittent, and water typically flows from it only after precipitation events. There may be some water seeping from the sand and gravel unit beneath the caprock and into Baker Spring. The area where Baker Spring is located is underlain by the Chinle clay. Deep infiltration of water is impeded by the low permeability of the clay. Therefore, seepage and/or precipitation/runoff into the Baker Spring area appear to be responsible for the intermittent localized flow and ponding of water in this area. Flows from this feature are intermittent, unlike those supplying the Wallach Concrete pits. This condition does not exist at the NEF site due to the absence of the caprock and the low permeability surface soils.

A recent investigation of the Baker Spring area supports the conclusion that the feature is man-made and results from the historical excavation of gravel and caprock materials that are present above the redbed clay. As a result of the excavation, Baker Spring is topographically lower than the surrounding area. Following rainfall events, ponding on the excavation floor occurs. Because the excavation floor consists of very low permeability clay of the redbed, limited vertical migration of the ponded water occurs. Shading from the high wall and trees that have flourished in the excavated area retard the natural evaporation rates and water stands in the pond for sometime. It is also suspected that during periods of ponding, surface water infiltrates into the sands at the base of the excavated wall and is retained as bank storage. As the surface water level declines, the bank storage is discharged back to the excavation floor.

A third instance of localized shallow groundwater occurrence exists to the east of the NEF site where several windmills on the WCS property were used to supply water for stock tanks. These windmills tapped small saturated lenses above the Chinle Formation red beds. The amount of groundwater in these zones is limited. The source of recharge for these localized perched zones is likely to be "buffalo wallows," (playas) depressions located near the windmills. The buffalo wallows are substantial surface depressions that collect surface water runoff. Water collecting in these depressions is inferred to infiltrate below the root zone due to the ponding conditions. WCS has drilled monitoring wells in these areas to characterize the nature and extent of the saturated conditions. Some of these wells are dry, owing to the localized nature of the perched conditions. When water is encountered in the sand and gravel above the Chinle

Formation red beds, its level is slow to recover following sampling events due to the low permeability of the perched saturated zones. The discontinuity of this saturated zone and its low permeability argue against its definition as an aquifer. No buffalo wallows or related groundwater conditions occur on or near the NEF site.

The hydrologic conditions that occur in the shallow surface regime at the NEF site are substantiated by field investigations including geochemical and soil-physics based techniques, as well as computer modeling, and show that there is no recharge occurring in thick, desert vadose zones with desert vegetation (Walvoord, 2002). Precipitation that infiltrates into the subsurface is efficiently transpired by the native vegetation. Vapor-phase movement of soil-moisture may occur, but it is also intercepted by the vegetation. In a thick vadose zone, such as at the NEF site, the deeper part of that zone has a natural thermal gradient that induces upward vapor diffusion. As a result, a small flux of water vapor rises from depth to the base of the root zone, and any infiltration coming from the land surface is captured by the roots of the plants within the top several meters of the profile. Effectively, there is a maximum negative pressure potential at the base of the root zone that acts like a sink, where water is taken up by the plants and transpired. These deep desert soil systems have functioned in this manner for thousands of years, essentially since the time of the last glacial period when precipitation rates fell dramatically. It is expected that these conditions will remain for several thousand more years (until the next glacial period), unless the hydrology and vegetation is altered dramatically.

3.2.4.3 Floods

The NEF site is located above the 100 or 500-year flood elevation (WBG, 1998 and FEMA, 1978).

The NEF site is contained within the Landreth-Monument Draw Watershed. The closest water conveyance is Monument Draw, a typically dry, intermittent stream located about 4 km (2.5 mi) west of the site. The maximum historical flow for Monument Draw is 36.2 m^3/s (1,280 ft^3/s) measured June 10, 1972. All other historical maximum measurements are below 2.0 m^3/s (70 ft^3/s) (USGS, 2003a). Therefore, a flood is not considered to be a design basis event for the NEF site.

3.2.4.4 Groundwater Hydrology

A subsurface investigation was performed for the NEF site during September 2003 to delineate specific hydrologic conditions. Figure 3.2-5 shows the locations of subsurface borings and observation wells.

The WCS facility, located east of the site in Texas, has had numerous subsurface investigations performed for the purpose of delineating and monitoring site subsurface hydrogeologic conditions. Much of this information is directly pertinent to the NEF site. The WCS hydrogeologic data was used in planning the recent NEF site investigations. A recent evaluation of potential groundwater impacts in the area provides a good overview of the investigations performed for the WCS facility. (Rainwater, 1996)

The NEF site investigation initiated in September 2003 had two main objectives: 1) to delineate the depth to the top of the Chinle Formation red beds to assess the potential for saturated conditions above the red beds, and 2) to complete three monitoring wells in the siltstone layer beneath the red beds to monitor water level and water quality within this thin horizon of perched intermittent saturation.

Nine boreholes oriented on a three-by-three grid were drilled to the top of the Chinle Formation red beds (Figure 3.2-5). Only one of the borings produced cuttings that were slightly moist at 1.8 to 4.2 m (6 to 14 ft) below ground surface; other cuttings were very dry. Left open for at least a day, no groundwater was observed to enter any of these holes. No samples could be collected for water quality analysis at the time of well construction. One groundwater sample has since been collected due to the limited groundwater occurrence.

The land surface elevation was surveyed at each of the nine borehole locations and the elevation of the top of the Chinle Formation red beds was computed. This information was combined with similar information from the WCS facility to produce an elevation map of the top of the red beds (See Figure 3.2-5). The dry nature of the soils from each of these borings supports a conclusion that there is no recharge from the ground surface at the site (Walvoord, 2002).

The three monitoring wells were installed at the end of September 2003. (Figure 3.2-5). Through the first month of monitoring only one well, MW-2, located at the northeast corner of the site, produced water. Several samples have been taken from that well.

Another factor to consider relative to hydrologic conditions at the NEF site is the presence of the Triassic Chinle Formation red bed clay. This clay unit is approximately 323 to 333 m (1,060 to 1,092 ft) thick beneath the site. With an estimated hydraulic conductivity on the order of 2.0 E-8 cm/s (7.9 E-9 in/s), the unit is very tight. This permeability is of the same order prescribed for engineered landfill liner materials. The expected vertical travel times through this clay unit would be on the order of thousands of years, based on this permeability and the thickness of the unit.

The first presence of saturated porous media beneath the site appears to be at the base of the Chinle red bed clay where there exists a low-permeability silty sandstone or siltstone. Borings and monitor wells at the WCS facility directly to the east of the NEF site have encountered this zone approximately 61 to 91 m (200 to 300 ft) below land surface. Wells completed in this unit are very slow to produce water. This makes sampling quite difficult. It is arguable whether this zone constitutes an aquifer, given the low permeability of the unit. As discussed above, three monitoring wells were installed on the NEF site in September 2003 with screened intervals within this siltstone unit. These wells are approximately 73 m (240 ft) deep. There is also a 30.5-m (100-foot) water-bearing sandstone layer at about 183 m (600 ft) below ground surface.

The first occurrence of a well-defined aquifer is approximately 340 m (1,115 ft) below land surface, within the Santa Rosa formation. Because of the depth below land surface to this unit, and the fact that the thick Chinle clay unit would limit any potential migration to depth, this aquifer has not been investigated. No impacts are expected to the Santa Rosa aquifer.

Based on groundwater levels in MW-2 and data from the adjacent WCS site, a groundwater gradient of 0.011 m/m (ft/ft) was determined, generally sloping towards the south. Hydraulic conductivity of the saturated layer, based on slug tests is estimated to be approximately 3.7 E-6 cm/s (1.5 E-6 in/yr). Based on the data collected at the NEF and WCS, the groundwater gradient in the siltstone unit at NEF is estimated to range from approximately 0.011 to 0.017 m/m (0.011 to 0.017 ft/ft).

Figure 3.2-6, Water and Oil Wells in the Vicinity of the NEF Site, is a map of wells and surface water features in the vicinity of the NEF site. The figure also includes oil wells. No water wells are located within 1.6 km (1 mi) of the site boundary.

3.2.4.5 Groundwater Chemistry

As discussed in Section 3.2.4.4, water resources in the area of the NEF site are minimal. Precipitation runoff at the site is effectively collected and contained by detention/retention basins and through evapotranspiration. It is highly unlikely that any groundwater recharge will occur at the site.

The first occurrence of groundwater beneath the NEF site is in a silty sandstone or siltstone horizon in the Chinle Formation, approximately 65 to 68 m (214 to 222 ft) below the surface. This unit is low in permeability and does not yield water readily. Groundwater quality in monitoring wells in the Chinle Formation, the shallowest saturated zone, is poor due to natural conditions. Samples from monitoring wells within this horizon on the WCS facility have routinely been analyzed with Total Dissolved Solids (TDS) concentrations between about 2,880 and 6,650 mg/l. Metal analyses from four background monitoring wells at the WCS site sampled during the period 1997-2000 show that essentially all results are below maximum contaminate limits (MCL) for EPA drinking water standards. The tightness of the formation, the limited thickness of saturation, and the poor water quality, support the argument that this zone does not constitute an aquifer.

Three monitor wells MW-1, MW-2, and MW-3, have been drilled and installed on the NEF site as shown on (Figure 3.2-5), and several water quality samples have been obtained. Water quality characteristics are similar to those for WCS site samples. A detailed discussion of the groundwater sample analysis is presented in Section 3.4.2, Water Quality Characteristics, of the Environmental Report.

3.2.5 Geology

This section identifies the geological, seismological, and geotechnical characteristics of the NEF site and its vicinity. Some areas immediately adjacent to the site have been thoroughly studied in recent years in preparation for construction of other facilities including the Waste Control Specialists (WCS) site and the former proposed Atomic Vapor Laser Isotope Separation (AVLIS) site. Data remain available from these investigations in the form of reports (WBG, 1998; TTUWRC, 2000). These documents and related materials provide a significant description of geological conditions for the NEF site. In addition, LES performed field investigations, where necessary, to confirm site-specific conditions.

3.2.5.1 Regional Geology

The site is located near the boundary between the Southern High Plains Section (Llano Estacado) of the Great Plains Province to the east and the Pecos Plains Section to the west. The boundary between the two sections is the Mescalero Escarpment, locally referred to as Mescalero Ridge. That ridge abruptly terminates at the far eastern edge of the Pecos Plains. The ridge is an irregular erosional topographic feature in southern Lea County where it exhibits relief of about 9 to 15 m (30 to 50 ft) compared with a nearly vertical cliff and relief of approximately 45 m (150 ft) in northwestern Lea County. The lower relief of the ridge in southeastern Lea County is due to partial cover by wind deposited sand (WBG, 1998). The dominant geologic feature of this region is the Permian Basin. The NEF site is located within the Central Basin Platform area. This platform occurs between the Midland and Delaware Basins, which comprises the Permian Basin. The basin, a 250 million-year-old feature, is the source of the region's prolific oil and gas reserves. The late Cretaceous to the early Tertiary (65 to 70 million years ago) marked the beginning of the Laramide Orogeny, which formed the Cordilleran Range to the west of the Permian Basin. That orogeny uplifted the region to its present elevation.

The primary difference between the Pecos Plains and the Southern High Plains physiographic sections is a change in topography. The High Plains is a large flat mesa which uniformly slopes to the southeast. In contrast, the Pecos Plains Section is characterized by its more irregular erosional topographic expression (WBG, 1998).

The Permian Basin, a massive subsurface bedrock structure, is a downward flexure of a large thickness of originally flat-lying, bedded, sedimentary rock. It dominates the geologic structure of the region. It extends to 4,880 m (16,000 ft) below msl. The NEF site is located above the Central Basin Platform that divides the Permian Basin into the Midland and Delaware sub-basins. The base of the Permian basin sediments extend about 1,525 m (5,000 ft) deep beneath the NEF site.

The top of the Permian deposits is approximately 434 m (1,425 ft) below ground surface. Overlying the Permian are the sedimentary rocks of the Triassic Age Dockum Group. The upper formation of the Dockum Group is the Chinle. Locally, the Chinle Formation consists of red, purple and greenish micaceous claystone and siltstone with interbedded fine-grained sandstone. The Chinle is regionally extensive with outcrops as far away as the Grand Canyon region in Arizona (WBG, 1998). Locally overlying the Chinle Formation in the Permian Basin is either the Tertiary Ogallala, Gatuña or Antlers Formations, or Quaternary alluvium. The Tertiary Ogallala Formation underlies all of the High Plains (to the east) and mantles several ridges in Lea County. Unconsolidated sediments northeast of the NEF site are recognized as the Ogallala and deposits west of the NEF site are mapped as the Gatuna or Antlers Formations. This sediment is described as alluvium (WBG, 1998) and is mined as sand and gravel in the NEF site.

The Chinle Formation is predominately red to purple moderately indurated claystone, which is highly impermeable (WBG, 1998). Red Bed Ridge is a significant topographic feature in this regional plain that is just north and northeast of the NEF site, and is capped by relatively resistant caliche. Ground surface elevation increases about 15 m (50 ft) from +1,045 m (+3,430 ft) to +1,059 m (+3,475 ft) across the ridge.

Recent deposits at the site and in the site area are primarily dune sands derived from Permian and Triassic rocks of the Permian Basin. The so-called Mescalero Sands cover approximately 80% of Lea County, locally as active sand dunes.

Two types of faulting were associated with early Permian deformation. Most of the faults were long, high-angle reverse faults with well over a hundred meters (several hundred feet) of vertical displacement that often involved the Precambrian basement rocks. The second type of faulting is found along the western margin of the platform where long strike-slip faults, with large displacements, are found. The nearest recent faulting to the site is defined by the New Mexico Bureau of Geology and Mineral Resources (NMIMT, 2003) and is over 161 km (100 mi) to the west associated with the deeper portions of the Permian Basin (Machette, 1998).

The large structural features of the Permian Basin are reflected only indirectly in the Mesozoic and Cenozoic rocks, as there has been virtually no tectonic movement within the basin since the Permian period. Figure 3.2-7, Permian Basin Geologic Structures and Profile, shows the structure that causes the draping of the Permian sediments over the Central Basin Platform structure, located approximately 2,134 m (7,000 ft) beneath the present land surface. The faults that uplifted the platform do not appear to displace the younger Permian sediments.

The Southeast New Mexico-West Texas area presently is structurally stable. The Permian Basin has subsided slightly since the Laramide Orogeny. This is believed to be a result of dissolution of the Permian evaporite layers by groundwater infiltration and possible from oil and gas extraction (WBG, 1998).

3.2.5.2 Site Geology

Topographic relief on the site is generally subdued. NEF site elevations range between about +1,033 and +1,045 m (+3,390 and +3,430 ft), mean sea level (msl) (See Figure 3.2-8, Site Topography). Finished site grade will range about +1,041 m (+3,415 ft), msl. The NEF site itself encompasses 220 ha (543 acres), of which 73 ha (180 acres) will be developed. Small-scale topographic features within the boundary of the proposed NEF site include a closed depression evident at the northern center of the site, the result of eolian processes, and a topographic high at the southwest corner of the site is created by dune sand. In general the site slopes from northeast to southwest with a general overall slope of about 0.5%. Red Bed Ridge (TTUWRC, 2000) is an escarpment of about 15 m (50 ft) in height that occurs just north and northeast of the NEF site. Geologically the site is located in an area where surface exposures consist mainly of Quaternary-aged eolian and piedmont sediments along the far eastern margin of the Pecos River Valley (NMIMT, 2003). Figure 3.2-9, Surficial Geologic Map of the NEF Site Area, is a portion of the Surficial Geologic Map of Southeast New Mexico (NMIMT, 1977), which includes the area of the NEF site. The surficial unit shown on this map at the NEF site is described as a sandy alluvium with subordinate amounts of gravel, silt and clay. Figure 3.2-9 also shows other surficial units in the site vicinity including caliche, a partly indurated zone of calcium carbonate accumulation formed in the upper layers of surficial deposits including tough slabby surface layers and subsurface nodules, fibers and veinlets; loose sand deposits, some gypsiferous, and subject to wind erosion. Other surficial deposits in the site area include floodplain channel deposits along dry channels and playa sands.

Recent deposits of dune sands are derived from Permian and Triassic rocks. These so-called Mescalero Sands (also known as the Blackwater Draw Formation) occur over 80% of Lea County and are generally described as fine to medium-grained and reddish brown in color. The USDA Soil Survey of Lea County identifies the dune sands at the site as the Brownsfield-Springer Association of reddish brown fine to loamy fine sands (USDA, 1974).

Figure 3.2-5 includes the NEF site and adjacent site borings and a geologic profile from the immediately adjacent parcel to the east that provides a representation of site geology. The profile shows alluvial deposits about 9 to 15 m (30 to 60 ft) thick, cemented by soft caliche layer 1 to 4 m (3 to 12 ft) that occurs at the top of the alluvium. Locally on the site dune sand overlies both these deposits. The alluvium rests on the red beds of the Chinle Formation, a silty clay with lenses of sandy clay or claystone and siltstone. Information from recent borings done on the NEF site is consistent with the data shown on Figure 3.2-5. Borings on the NEF site depicted on Figure 3.2-5 include:

- Three borings/monitoring wells (MW-1, MW2, and MW-3)
- Nine site groundwater exploration borings (B-1 through B-9)
- Five geotechnical borings (B-1 through B-5).

Other borings depicted on Figure 3.2-5, not on the NEF site, were performed by others.

The NEF site boring test records are shown on Figures 3.2-10 through 3.2-14. A key to the symbols and descriptions shown on the test records is provided in Figure 3.2-15, Soil Test Boring Key to Symbols and Descriptions.

The NEF site lies within the Landreth-Monument Draws Watershed. Site drainage is to the southwest with runoff not able to reach any water body before it evaporates. The only major regional drainage feature is Monument Draw, which is located just over 4 km (2.5 mi) west of the site, between the proposed NEF site and the city of Eunice, New Mexico (USDA, 1974). The draw begins with a southeasterly course to a point north of Eunice where it turns south and becomes a well defined cut approximately 9 m (30 ft) in depth and 550 to 610 m (1,800 to 2,000 ft) in width. The draw does not have through-going drainage and is partially filled with dune sand and alluvium.

Along Red Bed Ridge (TTUWRC, 2000), approximately 1.6 km (1.0 mi) northeast of the NEF site, is Baker Spring. The depression formed by Baker Spring contains water only intermittently.

No significant non-petroleum mineral deposits are known to exist in the vicinity of the NEF site. The surface cover of silty sand and gravel overlies a claystone of no economic value. No mineral operations are noted in Lea County by the New Mexico Bureau of Mines Inspection (NMBMI, 2001). Mining and potential mining of potash, a commonly extracted mineral in New Mexico, is followed by the New Mexico Energy, Minerals and Natural Resources Department, which maintains a map of areas with potash mines and mining potential (NMEMNRD, 2003). Those data indicate neither mining nor potential for mining of potash in the NEF site area.

The topographic quadrangle map that contains the site (USGS, 1979) contains 10 locations where sand and gravel have been mined from surface deposits, spread across the quadrangle, over an area about 12 by 14 km (7.5 by 8.9 mi), suggesting that suitable surficial deposits for borrow material are widespread.

Exploratory drill holes for oil and gas are absent from the site area and its vicinity, but are common 8 km (5 mi) west in and around the city of Eunice, New Mexico. That distribution, and the time period of exploration since the inception of exploration for this area, suggests that the potential for productive oil drilling at the NEF site is not significant.

Soil development in the region is generally limited due to its semi-arid climate. The site has a minor thickness of silty soil (generally less than 0.4 m (1.4 ft)) developed from subaerial weathering. Caliche deposits are common in the near-surface soils. A small deposit of active dune sand is present at the southwest corner of the site.

The U. S. Department of Agriculture soil survey for Lea County, New Mexico (USDA, 1974) categorizes site soils as hummocky loamy (silty) fine sand with moderately rapid permeability and slow runoff, well-drained non-calcareous loose sand, active dune sand and dune-associated sands. Near-surface caliche deposits may locally limit (limiting soil porosity) or enhance (fractured caliche) surface drainage. Figure 3.2-16, Site Soils Map, shows the soil map for the NEF site (USDA, 1974). The legend for that map lists each of the soils present at the NEF site describing them and along with their unified soil classification designations (ASTM, 1993).

3.2.5.3 Geotechnical Investigations

Previously completed geotechnical investigations on property near the site provide the following subsurface information. Based on the data from those investigations, subsurface conditions are described as follows. Topsoil occurs as 0.3 m (1 ft) or less of brown organic silty sand that overlies a formation of white or tan caliche. The caliche consists of very hard to friable cemented sand, conglomerate limestone rock, silty sand and gravel. A sand and gravel layer varying from 0 to 6 m (0 to 20 ft) in thickness occurs at the bottom of the caliche strata. Below the caliche is a reddish brown silt clay that extends to the termination of the borings, 30 to 91 m (100 to 300 ft) below grade. The red beds consist of a highly consolidated, impervious clay:

- mottled reddish brown-gray clay
- purple-gray silty clay and
- yellowish brown-gray silty clay
- siltstones and sandstone layers found at various depths with varying thicknesses.

The depth to the top of the red beds in borings done for engineering purposes ranged from about 3.6 to 9.1 m (12 to 30 ft).

The dry density of the clay ranges from 1.86 to 2.32 g/cm^3 (116 to 145 lb/ft^3), averaging 2.11 g/cm^3 (132 lb/ft^3). The red, reddish-brown or purple silty clays range in moisture content from 2.5% to 25%, averaging 8% 12% for most samples. Liquid limits for the clays range from 35% to 55% with plasticity indices ranging from 24 to 38. Percent passing the #200 sieve for the clays ranges from 87% to 99.8%.

The measured permeabilities for the reddish brown silty clays, sandstones and siltstones indicate the clay is highly impervious. The siltstones are slightly more permeable but still have relatively low permeability.

Unconfined compressive tests on the clay resulted in values of 136,000 kg/m^2 to 485,000 kg/m^2 (13.9 to 49.7 tons/ft^2) with an average value of 293,000 kg/m^2 (30 tons/ft^2).

A geotechnical investigation of the NEF site conducted in September 2003 consisted of 5 widely-spaced test borings that extended to depths of about 12 to 30.5 m (40 to 100 ft) using a hollow-stem auger and split-spoon sampling. Based on the boring results, up to 0.6 m (2 ft) of loose eolian sand underlain by dense to very dense, fine- to medium-grained sand and silty sand of the Gatuña/Antlers Formation was encountered. These sands are locally cemented with caliche deposits. Beneath the Gatuña/Antlers Formation is the Chinle claystone, a very hard highly plastic clay, which was encountered at depths of about 10.7 to 12.2 m (35 to 40 ft). One boring extended to 30.5 m (100 ft) deep and ended in the Chinle Formation. Blow-count N-values for about the top 7.6 m (25 ft) of sand and gravel ranged from about 20 to 76. Beneath that horizon the unit becomes denser or contains gravel to the extent that useful blow counts are not obtained. Where caliche cements the sand and gravel, N-values of over 60 are typical. Standard N-values were not available for samples in the underlying clay due to its hardness causing blow counts to range upwards of 100.

For samples from the shallow sand and gravel unit, California Bearing Ratio values of 10.5 and 34.4 were obtained along with a maximum dry density value of 1.97 g/cm^3 (123 lbs/ft^3). Fines in this material were generally non-plastic with 17% to 31% of samples finer than 200 sieve size. Clay samples had relatively high liquid limits of 50% to 60% and plastic limits of 18% to 23%, suggesting high silt content.

Footings bearing in the firm and dense sandy soils below the upper loose eolian soils are estimated to have an allowable bearing pressure of 34,177 kg/m^2 (7,000 lb/ft^3).

3.2.6 Seismology

The majority of earthquakes in the United States are located in the tectonically active western portion of the country. However, areas within New Mexico and the southwestern United States also experiences earthquakes, although at a lower rate and at lower intensities. Earthquakes in the region around the NEF site are isolated or occur in small clusters of low to moderate size events toward the Rio Grande Valley of New Mexico and in Texas, southeast of the NEF site.

3.2.6.1 Seismic History of the Region and Vicinity

The NEF site is located within the Permian Basin as shown on Figure 3.2-17, Tectonic Subdivisions of the Permian Basin (Talley, 1997). Specifically, the site is located near the northern end of the Central Basin Platform (CBP). The CBP became a distinct dividing feature within the Permian Basin as a result of Pennsylvanian and early Permian compressional stresses. This tectonism resulted in a deeper Delaware Basin to the west and shallower Midland Basin to the east of the ridge-like CBP.

The last episode of tectonic activity centered on the late Cretaceous and early Tertiary Laramide Orogeny that formed the Cordilleran Range to the west of the Permian Basin. The Permian Basin region was uplifted to its present position during this orogenic event. There has not been any further tectonic activity since the early Tertiary. Structurally, the Permian Basin has subsided slightly since the Larmaide tectonic event. Dissolution of Permian evaporate layers by groundwater infiltration or possibly from oil and gas extraction is suggested as a possible cause for this observed subsidence.

The 250 million year old Permian Basin is the source of abundant gas and oil reserves that continue to be extracted. These oil fields in southeast New Mexico are characterized as "in mature stage of secondary recovery effort" (Talley, 1997). Water flooding began in the late 1970's followed by CO_2 flooding now being used to enhance recovery in some fields. Industry case studies describe hydraulic fracturing procedures used in the Queen and San Andres formations near the NEF site that produced fracture half-lengths from 170 to 259 m (560 to 850 ft) in these formations.

Locations of recent tectonic faulting within the 322 km (200 mi) radius of the NEF site located in Lea County, New Mexico, were determined through literature research (DOE, 2003; Machette, 1998; Machette, 2000; USGS, 2004). No Quaternary faults are mapped for the site locale. The nearest recent faulting is situated more than 161 km (100 mi) west of the site (Machette, 1998). Figure 3.2-33, Quaternary Faults in New Mexico, and Figure 3.2-34, Quaternary Faults in Texas, illustrate traces of Quaternary Faults for New Mexico and adjacent areas of west Texas. The Quaternary geologic time period extends from 1.6 million years ago to the present. Other time sub-divisions within the Quaternary include the Late Quaternary that extends from 130,000 years ago to the present, and the Holocene, which includes the most recent 10,000-year time period.

Shown on Figures 3.2-33 and 3.2-34 are 1° Latitude by 2° Longitude geographic blocks. The NEF site is located in the Hobbs geographic block. Geographic blocks containing Quaternary faults are color-coded (i.e., non-gray). Figure 3.2-35, Quaternary Faults Within 322 km (200 mi) of NEF Site, shows geographic blocks for which Quaternary faults are mapped. All of these geographic blocks are located west of the NEF site. Figure 3.2-36, Locations of Nearest Faults to the NEF Site, shows the Quaternary fault locations detailed in the "Map and data for Quaternary faults and folds in New Mexico, U.S. Geological Survey (USGS) Open-File Report 98-521" (Machette, 2000). The block containing the site, as well as others due north, south, and east of the NEF site has no documented Quaternary faults. Quaternary faults within 322 km (200 mi) of the site are shown on Figure 3.2-35 using colored and numbered traces, and are plotted over shaded relief topographic maps. The use of topographic relief maps is highly illustrative, because ground deformations resulting from recent fault movements are usually manifested as prominent linear topographic features.

Figure 3.2-36 provides a summary of Quaternary fault locations, including fault names obtained from the "Map and data for Quaternary faults and folds in New Mexico, USGS Open-File Report 98-521" (Machette, 2000) and the "Earthquake Hazards Program, Quaternary Fault and Fold Database of the United States" (USGS, 2004).

Quaternary-Aged Faults designated as capable within 322 km (200 mi) of the NEF site include the West Delaware Mountain Fault Zone, the Guadalupe Fault, the East Sierra Diablo Fault, the East Flat Top Mountain Fault and the Alamogordo Fault at 185 km (115 mi), 191 km (119 mi), 196 km (122 mi), 200 km (124 mi) and 262 km (163 mi) from the site, respectively. In addition, the East Baylor Mountain – Carrizo Mountain Fault is located 201 km (125 mi) from the NEF and is considered a possible, capable fault, but movement within the last 35,000 years has not been demonstrated.

None of the capable faults pose a ground deformation hazard to the NEF site due to the distances (> 161 km (100 mi)) from the site, the northerly strike of these faults and the associated topographic landforms shown in Figure 3.2-36, Location of Nearest Faults to the NEF Site. The strikes of the assessed capable faults do not project toward the NEF site. Topographic features, like those correlated to the Quaternary faults west of the site, are not present near the NEF site, thus making it an unlikely scenario that unmapped, capable faults are located nearer than 161 km (100 mi) to the NEF site.

The study of historical seismicity includes earthquakes in the region of interest known from felt or damage records and from more recent instrumental records (since early 1960's). Most earthquakes in the region have left no observable surface fault rupture.

Figure 3.2-18, Seismicity Map for 200-Mile Radius of the NEF Site, indicates the location of earthquakes which have occurred within a 322 km (200 mi) radius of the NEF site with magnitude > 0. The earthquakes are also listed in Table 3.2-20, Location of Recorded Earthquakes Within a 322 km (200 mi) Radius of the NEF Site. Figure 3.2-19, Seismicity in the Immediate Vicinity of the NEF Site, indicates the location of earthquakes within about 97 km (60 mi) of the NEF site. Earthquakes, which have occurred within a 322 km (200 mi) radius of the NEF site with a magnitude of 3.0 and greater, are listed in Table 3.2-21, Earthquakes of Magnitude 3.0 and Greater Within 322 km (200 mi) Radius of the NEF Site.

The data reflected in the above figures and tables are from earthquake catalogs from the University of Texas Institute for Geophysics (UTIG, 2002), New Mexico Tech Historical Catalog (NMIMT, 2002), Advanced National Seismic System (USGS, 2003b) and the New Mexico Technical Regional Catalog, exclusive of Socorro New Mexico events (NMIMT, 2002).

Earthquake data for a 322 km (200 mi) radius of the NEF site were acquired from public domain resources. Table 3.2-22, Earthquake Data Sources for New Mexico and West Texas, lists organizations and data sources that were identified and earthquake catalogs that were obtained.

Earthquake parameters (e.g., date, time, location coordinates, magnitudes, etc.) from the data repositories listed in Table 3.2-22 were combined into a uniformly formatted database to allow statistical analyses and map display of the four catalogs. Through a process of comparison of earthquake entries among the four catalogs, duplicate events were purged to achieve a composite catalog. In addition, aftershocks and aftershock sequences were purged from one version of the catalog for computation of earthquake recurrence statistical models, which describe recurrence rates of earthquake main shocks. The composite list of earthquakes, with aftershock and aftershock sequences purged, for the 322 km (200 mi) radius of the NEF site is provided in Table 3.2-20. The regional seismicity map is shown on Figure 3.2-18. Local seismicity is shown on Figure 3.2-19, Seismicity in the Immediate Vicinity of the NEF Site. The large majority of events (i.e., 82%) in the composite catalog originate from the Earthquake Catalogs for New Mexico (exclusive of the Socorro New Mexico immediate area) (NMIMT, 2002) as observed in the event counts in Table 3.2-22. Earthquake magnitudes in these catalogs (NMIMT, 2002) are tied to the New Mexico duration magnitude scale, M_d, that in turn approximate Local Magnitude, M_L. All events in the composite catalog are specified to have an undifferentiated local magnitude.

Table 3.2-21 shows all earthquake main shocks of magnitude 3.0 and larger within a 322 km (200 mi) radius of the NEF site. The largest earthquake within 322 km (200 mi) of the NEF is the August 16, 1931 earthquake located near Valentine, Texas. This earthquake has an estimated magnitude of 6.0 to 6.4 and produced a maximum epicentral intensity of VIII on the Modified Mercalli Intensity (MMI) Scale. The intensity observed at the NEF site is IV on the MMI scale (NMGS, 1976). A copy of the MMI scale is provided in Table 3.2-23, Modified Mercalli Intensity Scale.

The closest of these moderate earthquakes occurred about 16 km (10 mi) southwest of the site on January 2, 1992.

It is noted that the University of Texas Geophysics Institute Catalog of West Texas Earthquakes reports a smaller magnitude of 4.6 and a more easterly epicenter location in Texas. Table 3.2-24, Comparison of Parameters for the January 2, 1992 Eunice, New Mexico Earthquake, shows the location and size parameters for the Earthquake. Parameters given by New Mexico Tech Regional Catalog were adopted for the seismic hazard assessment of the NEF site.

3.2.6.2 Correlation of Seismicity with Tectonic Features

Earthquake epicenters scaled to magnitude for the site region are plotted over Permian Basin tectonic elements on Figure 3.2-20, Regional Seismicity and Tectonic Elements of the Permian Basin. Most epicenters lie within the Central Basin Platform, however, earthquake clusters also occur within the Delaware and Midland Basins. Although events local to the NEF site are likely induced by gas/oil recovery methods, the resulting ground motions are transmitted similar to earthquakes on tectonic faults and impacts at the NEF site are analyzed using standard seismic hazard methods. Furthermore, given the published uncertainties on discrimination between natural and induced seismic events and that earthquake focal depths, critical for correlation with oil/gas reservoirs, are largely unavailable, the January 2, 1992 event is attributed to a tectonic origin. For this magnitude 5 earthquake, focal depths range from 5 km (3.1 mi) (USGS, 2004) to 12 km (7.5 mi) (DOE, 2003). Therefore, studies conclude that seismological data are insufficient for this moderate earthquake to constrain the depth sufficiently to permit a correlation with local oil/gas producing horizons.

Analysis of the spatial density of earthquakes in the composite catalog is shown on Figure 3.2-21, Earthquake Frequency Contours and Tectonic Elements of the Permian Basin. This form of spatial analysis has historically been used to define the geometry of seismic source zones for seismic hazard investigations (USGS, 1997; USGS, 1976a). Seismic source areas for the NEF site region are determined on the basis of the earthquake frequency pattern shown on Figure 3.2-22, Seismic Source Areas for Earthquake Frequency Statistical Analyses. The NEF site is located near the northern end of the region of highest observed earthquake frequency within the CBP of the Permian Basin.

The Waste Isolation Pilot Plant (WIPP) Safety Analysis Report (SAR) (DOE, 2003) suggests that the cluster of small events located along the CBP (Figure 3.2-20) are not tectonic in origin, but are instead related to water injection and withdrawal for secondary recovery operations in oil fields in the CBP area. Such a mechanism for the CBP seismic activity could provide a reason why the CBP is separable from the rest of the Permian Basin on the basis of seismicity data but not by using other common indicators of tectonic character. Both the spatial and temporal association of CBP seismicity with secondary recovery projects at oil fields in the area are suggestive of some cause and effect relationship of this type.

3.2.6.3 Earthquake Recurrence Models

Earthquake recurrence models describe the exponential frequency versus magnitude behavior observed for earthquake activity (Gutenberg, 1944). The exponential recurrence model is commonly shown as Equation [3.2-1].

$$\mathrm{Log}_{10} N_C = a + b(M) \qquad \text{[Eq. 3.2-1]}$$

Where: N_C = cumulative number per time duration (i.e., per year)
a = a-value, indicator of activity rate
b(M) = b-value, with negative slope due to observation that smaller magnitude events occur more frequently than larger magnitude events. Typical range of b-values is -0.5 to -1.5, normally closer to -1.0.

Earthquake recurrence models were computed for the entire 322 km (200 mi) radius composite catalog and for two smaller regions. The smaller regions are defined by patterns of seismic activity as noted at closer distances to the site. Region 1 shown on Figure 3.2-22 includes clusters of earthquakes within an approximate 161 km (100 mi) radius of the site. The second sub-region includes the high-density earthquake pattern observed in the CBP. A tectonic origin for all events in the CBP was conservatively assumed.

Results of statistical analyses performed on the 322 km (200 mi) composite catalog and two sub-regions are illustrated on Figures 3.2-23 through 3.2-25. Best fit models and models for which the b-value is constrained to a value of -0.9 were computed. These models are numerically compared in Table 3.2-25, Earthquake Recurrence Models for the NEF Site Region.

Earthquake recurrence models provided in the WIPP SAR (DOE, 2003) for more distant seismic zones including the two Rio Grande Rift source zone alternatives (see Figure 3.2-26, Alternate Seismic Source Geometries Used in the WIPP Seismic Hazard Study) were used in the hazard assessment of the NEF site. Recurrence models from the WIPP SAR (DOE, 2003) are shown in Table 3.2-32, Horizontal Response Spectrum for the 10,000-Year Design Earthquake. Preparers of the WIPP SAR (DOE, 2003) expressed an opinion that magnitudes in the available earthquake catalog (pre-1983) were underestimated. Therefore, two models were used to address this magnitude scaling issue. The model for corrected magnitude raised the a-value in the recurrence models by 0.5 units. Both the magnitude-corrected and uncorrected recurrence models are listed in Table 3.2-26, Earthquake Recurrence Models for the CBP in the WIPP SAR.

3.2.6.4 Probabilistic Seismic Hazard Analysis

3.2.6.4.1 Ground Motion Attenuation Models

A site-specific probabilistic seismic hazard analysis was performed for the NEF site using the seismic source zone geometries shown on Figures 3.2-22 and 3.2-26 and earthquake recurrence models listed in Tables 3.2-25 and 3.2-26. Seismic hazard computations were performed using the EQRISK computer program (Cornell, 1968; USGS, 1976b).

In addition to seismic source zones and earthquake recurrence models, computations of probabilistic seismic hazard require ground motion attenuation models suited for the regional and local seismic wave transmission characteristics. Two attenuation models were used in the analysis. The WIPP SAR (DOE, 2003) selected an attenuation model developed by O.W. Nuttli (US Army WES, 1973) for application in the central United States. This model was selected due to the precedence of its usage in the WIPP SAR seismic hazard assessment, and to its conservative predictions compared to other published models. This ground acceleration model is given in Equation 3.2-2.

$$Ln(a) = 2.833 + 0.92(M_L) - 1.0(Ln(R)) \qquad \text{[Eq. 3.2-2]}$$

Where:
a = horizontal ground acceleration in cm/s^2 units
M_L = Local Magnitude
R = distance from the earthquake focus to the site

Sensitivity to the attenuation model was studied by calculating seismic hazard curves for an attenuation model that approximates the Toro peak ground acceleration model (Toro, 1997). This model is provided in Equation 3.2-3 and is illustrated on Figure 3.2-27, Comparison of PGA Attenuation for a Magnitude 5.0 Earthquake.

$$Ln(a) = 2.80 + 0.92(M_L) - 1.05(Ln(R)) - 0.003(R) \qquad \text{[Eq. 3.2-3]}$$

Where:
a = horizontal ground acceleration in cm/s^2 units
M_L = Local Magnitude
R = distance from the earthquake focus to the site

It is noted that the Toro attenuation model provides coefficients for magnitudes scaled to the Lg-phase, m_{bLg}, and for Moment magnitude, M_O. Due to the magnitude scaling of events in the composite catalog, the moment magnitude scaling is preferred to Lg magnitude scaling for the

Toro model. In addition, the Toro model has a more sophisticated functional form that flattens the PGA predictions at distances less than 10 km (6.2 mi).

In addition, probabilistic response spectra (i.e. uniform hazard response spectra) are computed for the NEF site using the Nuttli spectral attenuation models (Nuttli, 1986) listed in Table 3.2-27, Attenuation Model Formulas and Coefficients. The Nuttli spectral velocity attenuation models are considered to predict ground motions at "firm rock" conditions, which is the rock condition attributed to the Triassic Age claystones underlying the NEF site. For comparative purposes, the Nuttli (Nuttli, 1986), Toro (Toro, 1997) and WIPP SAR Nuttli (US Army WES, 1973) attenuation models are plotted on Figure 3.2-21 along with the McGuire (EPRI, 1988) attenuation model and the approximation of the Toro attenuation models.

3.2.6.4.2 Probabilistic Seismic Hazard Results

Total seismic ground motion hazard to a site results from summation of ground motion effects from all distant and local seismically active areas. The contribution to total hazard at the NEF site from more distant seismic activity in the Rio Grande Rift zones is examined first. As noted above, seismic source zone geometries (Figure 3.2-26) and recurrence rates (Table 3.2-26) were taken directly from the WIPP SAR (DOE, 2003). Recurrence rates for the magnitude corrected, and magnitude uncorrected recurrence models were used in the hazard calculations. This recurrence model variation coupled with two seismic source zone geometries results in four seismic hazard curves. In addition, maximum magnitudes of 7.8 for the Rio Grande Rift (DOE, 2003) were used for this hazard calculation. Peak ground acceleration seismic hazard results at the NEF site from the Rio Grande Rift source zone alternatives are listed in Table 3.2-28, Seismic Hazard Results at NEF Site From Rio Grande Rift Seismic Source Zones. These hazard results are plotted on Figure 3.2-28, Seismic Hazard at the NEF Site From Rio Grande Rift Seismic Sources. Seismic hazard curves shown on Figure 3.2-28 are annotated to identify the 250-year, 475-year and 10,000-year earthquake levels. It is noted that the 475-year event in most cases is strictly defined as the event with a 10% probability of being exceeded in 50 years. Strict maintenance of this probability in 50-years equates to an annual probability of 0.0021 of exceeding a 0.10 g peak horizontal acceleration and a return period of 475-years.

Seismic hazard results for the NEF site due to seismic activity in local seismic zones (i.e. seismic zones that contain the site) are listed in Table 3.2-29, Seismic Hazard Results at NEF Site From Local Source Zones. Seismic hazard curves are plotted on Figure 3.2-29, Seismic Hazard at the NEF Site From Local Seismic Zone Sources. Local seismic zones include those geometries shown on Figure 3.2-22. The largest zone includes the 322 km (200 mi) radius of the NEF site for which earthquake data were assembled. The largest earthquake contained in this 322 km (200 mi) zone is the 1931 Valentine, Texas, event with an estimated magnitude of 6.0 to 6.4. Alternative maximum magnitudes, M_X, of 6.5 and 6.0 are assigned to this 322 km (200 mi) region for seismic hazard computations.

The alternative local seismic source zone geometry is defined within a more limited site radius of 161 km (100 mi). Embedded within this 161 km (100 mi) zone is the sub-region defined by the enhanced density of earthquake epicenters centered on the CBP (see Figure 3.2-21 and Figure 3.2-22). The maximum historical earthquake within these zones is the January 2, 1992, earthquake. A maximum magnitude of 6.0 is used for computation of seismic hazard curves. An identical maximum magnitude of 6.0 was specified in the WIPP SAR (DOE, 2003) for its CBP seismic source zone alternatives. In addition, the WIPP study used a smaller maximum magnitude of 5.0 in their hazard analysis due to the lack of recent geologic evidence of tectonism and likely association of events with secondary oil/gas recovery efforts in this area. Sensitivity to the maximum magnitude parameter is examined by computing seismic hazard curves for M_X set to 6.0 as well as to 5.25 for the 161 km (100 mi) zone and the CBP embedded zone. Seismic hazard results shown in Table 3.2-29 and on Figure 3.2-29, illustrate the various sensitivities to choices of seismic source zones, attenuation models and maximum magnitudes, M_X.

Figure 3.2-30, Zoom of Seismic Hazard at the NEF Site From Local Seismic Zone Sources, provides a zoomed-in view of the calculated seismic hazard curves for the NEF site. Table 3.2-30, Peak Acceleration Seismic Hazard Summary for the NEF Site, provides an interpretation of these hazard curves for the 250-year and 475-year earthquake levels.

Total seismic ground motion hazard to a site results from summation of ground motion effects from all distant and local seismically active areas. A total of 12 seismic hazard curves were developed for a combination of various source zones, attenuation models, b-values and upper bound magnitudes. For the purpose of selecting the characteristic peak ground acceleration associated with specific return periods, a resultant seismic hazard curve was developed through a weighted average of the individual curves. The seismic hazard curves and weighted average hazard result are shown in Figure 3.2-29 and Figure 3.2-30.

The 250-year and 475-year return period peak horizontal ground accelerations are estimated at 0.024 g and 0.036 g, respectively. The 10,000-year return period peak horizontal ground acceleration is estimated at 0.15 g. This return period is equivalent to a mean annual probability of 1.0 E-4.

Since it is currently not possible to definitively differentiate natural tectonic from induced seismic events in the study region, the probabilistic seismic hazard estimates for the NEF site assumed a tectonic origin for all events in the CBP sub-region. However, for cases of uncertainty, sensitivity analyses provide valuable insights into the impacts of induced earthquakes on the seismic hazard analysis. The following sensitivity analysis results are provided to show trends in seismic hazard results for assumptions that increasing percentages of earthquakes in the CPB seismic source zone are induced by oil/gas recovery activities.

Two hypotheses are considered in the seismic hazard sensitivity analyses. First is the case that a fraction of earthquakes of all magnitudes are induced. Second is the case that only smaller magnitude earthquakes (e.g., less than M=3.5) are likely induced while larger events result from tectonic processes. For the first case, the hypothesis is that a large fraction of events in the CBP was induced by oil/gas recovery efforts, is modeled by scaling the CBP recurrence model by factors of 0.15, 0.5, and 0.85. These scaling factors are applied to the entire recurrence model such that the predicted frequencies of events for all magnitudes are scaled by these factors. The three scaling factors are used to model the general commentary that a "large fraction" of CPB events are induced. For the second case, the concept that many of the small events could be induced while larger events have tectonic origins is modeled by re-computation of the recurrence model for the CPB following removal of 50% of events with magnitudes less than 3.5. This second case results in a recurrence model that predicts relatively fewer small magnitude events, and recurrence rate of larger events of magnitude 5.0 and greater remains unchanged.

Seismic hazard sensitivity results only show a significant impact when a scaling factor of 0.15 is applied to the total recurrence model. For this case, peak horizontal acceleration is reduced from about 0.15 g to about 0.10 g at 1.0 E-4 annual exceedance probability. Application of a scaling factor of 0.50 to the entire model resulted in a peak horizontal acceleration near 0.13 g at 1.0 E-4 annual exceedance probability. Two of the cases, scaling the entire recurrence model by 0.85, and determination of a new model based on removal of 50% of events smaller than M=3.5, showed little sensitivity. Given uncertainties related to the tectonic vs. induced nature of larger regional events, and high likelihood that many smaller events are induced by ongoing oil/gas recovery activities, results of the last sensitivity analysis (e.g. removal of smaller events only) are preferred. The negligible sensitivity to removal of smaller events emphasizes that seismic hazard in large part is determined by the assessed regional frequency of events with magnitudes larger than 5.0.

3.2.6.4.3 Uniform Hazard Response Spectra

Probabilistic ground motion response spectra are derived for the NEF site using a combination of the Nuttli spectral attenuation model (Nuttli, 1986) and appropriate soil amplification factors currently used in Seismic Building Code applications. The Nuttli spectral velocity attenuation models are considered to predict ground motions at "firm rock" conditions, which is the rock condition attributed to the Triassic Age claystones underlying the NEF site. Descriptive characterization of the site surficial material composition and thickness supports a site soil classification of C. This site class (Dobry, 2000) accommodates gravelly soils underlain by soft rocks, which appear to be present at the site. Soil amplification factors for Site Class C include:

For $S_s < 0.25$; short period site amplification factor, $F_a = 1.2$
For $S_l < 0.10$; long period site amplification factor, $F_v = 1.7$
Where S_s and S_l are short and long period
rock acceleration levels, respectively.

Horizontal component bedrock and ground surface response spectra (five percent damping ratio) for soil profile type C for the 10,000-year earthquake are plotted on Figure 3.2-31, Horizontal Response Spectra for the 10,000-Year Earthquake, Bedrock and Soil Class C for the NEF Site. By definition of their calculation, these response spectra have an equal probability of 0.005% of being exceeded in 50 years at each period in the range of 0.02 to 2.0 s.

Horizontal and vertical component uniform hazard response spectra (five percent damping) for the 10,000-year earthquake at ground surface for Soil Class C are plotted on Figure 3.2-32. Vertical component earthquake response spectra are recommended in NRC Regulatory Guide 1.60 (NRC, 1973) to be determined as a function of frequency. Table 3.2-31, Regulatory Guide 1.60 Ratio of Vertical to Horizontal Component Design Response Spectra, summarizes the ratio of vertical and horizontal component earthquake response spectra.

The vertical component 10,000-year response spectrum was determined using the formulation shown in Table 3.2-31.

Numerical values for the 10,000-year horizontal and vertical design response spectra for five percent damping are listed in Table 3.2-32, Horizontal Response Spectrum for the 10,000-Year Design Earthquake, and Table 3.2-33, Vertical Response Spectrum for the 10,000-Year Design Earthquake, respectively.

3.2.6.5 Selection of the Design Basis Earthquake

While conducting the Integrated Safety Analysis (ISA), an unmitigated accident due to a seismic event was assumed to result in high public consequences. Therefore, the likelihood of the event (seismically-induced high public consequences) needs to be "highly unlikely." In accordance with NUREG-1520 (NRC, 2002), for the NEF this equates to a probability of occurrence of less than 1.0 E-5 per year.

To define the design basis earthquake (DBE), information from DOE Standard DOE-STD-1020-2002 (DOE, 2002) and ASCE Standard Seismic Design Criteria (ASCE, 2003) was considered along with the results of the seismic portion of the ISA and the site-specific probabilistic seismic hazard analysis performed for the NEF site.

The DOE and ASCE standards outline a methodology to demonstrate compliance to a target performance goal of 1.0 E-5 annual probability by designing to a seismic hazard of 1.0 E-4 annual probability. The difference between the design level and the performance target is accounted for in the detailed design process by confirmatory calculations.

Based on these approaches, the DBE for the NEF has been selected as the 10,000-year (1.0 E-4 mean annual probability) earthquake. For the NEF, following the DOE or ASCE approach provides a risk reduction ratio of design to target performance of 10 (1.0 E-4/1.0 E-5). This DBE will be used in the detailed design process to demonstrate compliance with the overall ISA performance requirements. This will be accomplished by confirmatory seismic performance calculations for the seismic Items Relied on for Safety (IROFS) during detailed design. The DOE and ASCE standards address design and evaluation of structures, systems, and components (SSCs). The equivalents of SSCs for the NEF are considered to be the IROFS and the items that may affect the function of IROFS. The objective of the NEF seismic design approach is to demonstrate that use of this DBE achieves a likelihood of unacceptable performance of less than approximately 1.0 E-5 per year, by introducing sufficient design safety margins, i.e., conservatism, during the design process to allow for demonstration of compliance to the target performance goal. The DOE and ASCE standards implement this objective using slightly different methodologies with the same end result, i.e., demonstration of compliance to the target performance goal.

In the DOE approach, the deterministic seismic evaluation and acceptance criteria are structured to achieve less than a 10% probability of unacceptable performance for a SSC subjected to the scaled design/evaluation basis earthquake (SDBE). The SDBE is defined in the DOE approach as the product of the DBE times a factor of 1.5 and a scale factor, which is a function of the slope of the seismic hazard curve.

The ASCE approach is based on achieving the target performance goal annual frequencies by incorporating sufficient conservatism in the seismic demand and structural capacity evaluations to achieve both of the following:

- Less than about a 1% probability of unacceptable performance for the DBE ground motion
- Less than a 10% probability of unacceptable performance for a ground motion equal to 150% of the DBE ground motion

The ASCE method is based on achieving both of the above probability goals, which represent two points on the underlying fragility curve. Meeting these two probability goals allows the target performance probabilities to be achieved with less possibility of non-conservatism. The resulting nominal factors of safety against conditional probability of failure are 1.0 and 1.5, respectively, for the above two goals.

The actual seismic design detailed approach for NEF will be based on the DOE or ASCE method and finalized prior to detailed design. The safety margins will be representative of those discussed above and described in more detail in the DOE and ASCE standards.

The difference between the mean annual probabilities for design (1.0 E-4) and performance (1.0 E-5) is achieved through conservatism in the design (factors of safety), elasticity in the structures, and conservatism in the evaluation of the design.

The design response spectra, horizontal and vertical, are based on the 10,000-year uniform hazard response spectra described in Section 3.2.6.4.3, Uniform Hazard Response Spectra. The soil amplification factors described in Section 3.2.6.4.3 will be verified during the detailed design phase of the NEF project.

3.2.7 Stability of Subsurface Materials

A geotechnical investigation of the site conducted in September 2003 consisted of 5 widely-spaced test borings that extended to depths of about 12 to 30.5 m (40 to 100 ft) using a hollow-stem auger and split-spoon sampling. Based on the boring results, up to 0.6 m (2 ft) of loose eolian sand underlain by dense to very dense, fine- to medium-grained sand and silty sand of the Gatuña/Antlers Formation was encountered. These sands are locally cemented with caliche deposits. Beneath the Gatuña/Antlers Formation is the Chinle claystone, a very hard highly plastic clay, which was encountered at depths of about 10.7 to 12.2 m (35 to 40 ft). One boring extended to 30.5 m (100 ft) deep and ended in the Chinle Formation. Blow-count N-values for about the top 7.6 m (25 ft) of sand and gravel ranged from about 20 to 76. Beneath that horizon the unit becomes denser or contains gravel to the extent that useful blow counts are not obtained. Where caliche cements the sand and gravel, N-values of over 60 are typical. Standard N-values were not available for samples in the underlying clay due to its hardness causing blow counts to range upwards of 100.

For samples from the shallow sand and gravel unit, California Bearing Ratio values of 10.5 and 34.4 were obtained along with a maximum dry density value of 1.97 g/cm^3 (123 lbs/ft^3). Fines in this material were generally non-plastic with 17% to 31% of samples finer than 200 sieve size. Clay samples had relatively high liquid limits of 50% to 60% and plastic limits of 18% to 23%, suggesting high silt content.

Footings bearing in the firm and dense sandy soils below the upper loose eolian soils are estimated to have an allowable bearing pressure of 34,177 kg/m^2 (7,000 lbs/ft^3).

The five borings are not sufficient to adequately define subsurface conditions for final design purposes, but they are acceptable for judging the feasibility of developing the site. Assuming that the borings are generally representative of subsurface conditions, the site is considered acceptable for the facility features supported on a system of shallow foundations.

The surface deposits silty sands will be removed to expose the more firm soil structures. In this case, footings bearing in the firm and dense sandy soils below the upper, loose eolian soils can be designed for an allowable bearing pressure of 34,000 kg/m^2 (7,000 lb/ft^2). Due consideration will be given to settlement and differential settlement during final design.

To support the final design of the NEF, additional soil borings will be collected from the NEF site. Laboratory testing will be performed on soil samples and additional in-situ testing will be performed to determine static and dynamic soil properties. Using the soil information obtained, the following activities will be conducted.

- The assessment of soil liquefaction potential will be performed using the applicable guidance of Regulatory Guide 1.198, Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites, dated November 2003 (NRC, 2003).
- Allowable bearing pressures provided in the ISA Summary will be confirmed using the applicable methods of Naval Facilities Engineering Command Design Manual NAVFAC DM-7.02, Foundations and Earth Structures, dated 1986 (NAVFAC, 1986a); Foundation Engineering Handbook, H.F. Winterkorn and H.Y. Fang, dated 1975 (Winterkorn, 1975); and Foundation Analysis and Design, J.E. Bowles, dated 1996 (Bowles, 1996).
- Building settlement analysis will be performed using the applicable methods of NAVFAC DM-7.01, Soil Mechanics, dated 1986 (NAVFAC, 1986b); and Foundation Engineering Handbook, H.F. Winterkorn and H.Y. Fang, dated 1975 (Winterkorn, 1975). The acceptance criteria for the building settlement analysis will be based on Urenco design criteria for allowable total and differential settlement of equipment and buildings.

3.2.7.1 Liquefaction Susceptibility

Liquefaction potential is greatest where the groundwater level is shallow; and submerged, loose fine sands occur within a depth of about 15 m (50 ft). Liquefaction potential decreases as grain size and clay and gravel content increase.

The soils at the site are dense to very dense. Groundwater was encountered in the site soil borings drilled to a depth of more than 30 m (100 ft) below the ground surface. The nature of the soils and the absence of groundwater near the surface would make the potential for liquefaction remote.

3.2.8 References

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TABLES

Table 3.2-1 Population and Population Projections, 1970-2040

Topic	Area				
	Lea County, NM	**Andrews County, TX**	**Lea-Andrews Combined**	**New Mexico**	**Texas**
Population/Projected Growth					
1970	49,554	10,372	59,926	1,017,055	11,198,567
1980	55,993	13,323	69,316	1,303,303	14,225,512
1990	55,765	14,338	70,103	1,515,069	16,986,510
2000	55,511	13,004	68,515	1,819,046	20,851,820
2010	60,702	15,572	76,274	2,091,675	23,812,815
2020	62,679	16,497	79,176	2,358,278	26,991,548
2030	64,655	17,423	82,078	2,624,881	30,170,281
2040	66,631	18,348	84,979	2,891,483	33,349,013
Percent Change					
1970-1980	13.0	28.5	15.7	28.1	27.0
1980-1990	-0.4	7.6	1.1	16.2	19.4
1990-2000	-0.5	-9.3	-2.3	20.1	22.8
2000-2010	9.4	19.7	11.3	15.0	14.2
2010-2020	3.3	5.9	3.8	12.7	13.3
2020-2030	3.2	5.6	3.7	11.3	11.8
2030-2040	3.1	5.3	3.5	10.2	10.5

Source: U. S. Census Bureau

Table 3.2-2 Educational Facilities Near the Site
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School	Grades	Distance km (mi)	Direction	Population	Student-Teacher Ratio
Lea County, New Mexico					
Eunice High School	9-12	8.6 (5.3)	W	207	16:1
Caton Middle School	6-8	8.6 (5.3)	W	128	15:1
Mettie Jordan Elementary School	DD, K-5	8.6 (5.3)	W	269	21:1
Eunice Holiness Academy	1-12	8.2 (5.1)	W	14	6:1

Note: DD = Development Delayed Class

Source: Eunice School District
National Center for Educational Statistics
U.S. Census Bureau

Table 3.2-3 Land Use Within 8 km (5 mi) of the Site
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Classification	Area							Description
	(Hectares)			(Acres)			Percent	
	New Mexico	Texas	Total	New Mexico	Texas	Total		
Built Up	243	0	243	601	0	601	1.2	Residential; industrial; commercial services
Rangeland	12,714	7,213	19,927	31,415	17,823	49,238	98.5	Herbaceous rangeland; shrub and brush rangeland; mixed rangeland
Barren	69	0	69	170	0	170	0.3	Bare exposed rock; transitional areas; beaches; sandy areas other than beaches
Total	13,026	7,213	20,239	32,186	17,823	50,009	100.0	

Table 3.2-4 Agriculture Census, Crop, and Livestock Information
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Information	County			
	Lea (New Mexico)		Andrews (Texas)	
Census Data (1992 & 1997)	1997	1992	1997	1992
Number of Farms	528	544	142	134
Total Land in Farms ha (acres)	810,161 (2,001,931)	869,861 (2,149,450)	335,431 (828,859)	389,545 (962,576)
Avg. Farm Size ha (acres)[1]	1,535 (3,792)	1,599 (3,951)	2,362 (5,837)	2,907 (7,183)
Crop Annual Average Yields (Most Current)	Area Harvested Hectares (Acres) in 2001	Yield per Hectare (Acre) in 2001	Area Harvested Hectares (Acres) in 2002	Yield per Unit Area in 2001
Chili Peppers	324 (800)	4.49 MT/ha (2.0 tons/acre)	0	0
Wheat	3,035 (7,500)	3.91 m^3/ha (45.0 bu/acre)	81 (200)	2.61 m^3/ha (30 bu/acre)
Grain Sorghum	688 (1,700)	3.66 m^3/ha (42.1 bu/acre)	688 (1,700)	1,384 kg/ha (1,235 lb/acre)
Peanuts	5,828 (14,400)	3,182 kg/ha (2,840 lb/acre)	2,266 (5,600)	4,521 kg/ha (4,035 lb/acre)
All Hay	4,047 (10,000)	10.9 MT/ha (4.72 tons/acre)	0	0
Alfalfa Hay	2,428 (6,000)	13.6 MT/ha (6.0 tons/acre)	0	0
Pecans[2]	213 (526)	0	0	0
Upland Cotton	8,984 (22,200)	703 kg/ha (627 lb/acre)	7,811 (19,300)	435 kg/ha (388 lb/acre)

Table 3.2-4 Agriculture Census, Crop, and Livestock Information
Page 2 of 2

Information	County	
	Lea (New Mexico)	Andrews (Texas)
Livestock (Most Current)	Number in 2001	Number in 2002
All Cattle	82,000	13,000
Beef Cows	27,000	6,000
Milk Cows	25,000	0
Other Cattle (includes cattle on feed)	30,000	0
Sheep and Lambs	4,000	0

1 Average Value per ha (acre) [1998]: New Mexico $536 ($217)/Texas $1,465 ($593) (USDA, National Agricultural Statistical Service)

2 1997 Census Data

Table 3.2-5 Midland-Odessa, Texas, Wind Data
1961-1990
Page 1 of 1

	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sept	Oct	Nov	Dec	Year
Mean Speed m/sec (mi/hr)	4.6 (10.4)	5.0 (11.2)	5.5 (12.4)	5.6 (12.6)	5.5 (12.4)	5.5 (12.2)	4.8 (10.7)	4.4 (9.9)	4.4 (9.9)	4.4 (9.9)	4.6 (10.3)	4.5 (10.1)	4.9 (11.0)
Prevailing Direction degrees from True North	180	180	180	180	180	160	160	160	160	180	180	180	180
Max 5-second speed m/sec (mi/hr)	22.8 (51.0)	23.2 (52.0)	24.1 (54.0)	26.4 (59.0)	24.6 (55.0)	21.9 (49.0)	26.4 (59.0)	28.6 (64.0)	31.3 (70.0)	20.6 (46.0)	20.1 (45.0)	21.9 (49.0)	31.3 (70.0)

Local Climatological Data Annual Summary with Comparative Data for Midland-Odessa, Texas, National Oceanic and Atmospheric Administration, 2002.

Table 3.2-6 Roswell, New Mexico, Wind Data
1961-1990
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	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sept	Oct	Nov	Dec	Year
Mean Speed m/sec (mi/hr)	3.1 (6.9)	3.6 (8.1)	4.2 (9.5)	4.4 (9.8)	4.3 (9.6)	4.3 (9.6)	3.8 (8.5)	3.4 (7.7)	3.4 (7.6)	3.3 (7.3)	3.2 (7.2)	3.1 (6.9)	3.7 (8.2)
Prevailing Direction degrees from True North	360	160	160	160	160	160	140	140	160	160	160	360	160
Max 5-second speed m/sec (mi/hr)	24.1 (54.0)	24.1 (54.0)	24.1 (54.0)	26.4 (59.0)	24.6 (55.0)	27.7 (62.0)	26.4 (59.0)	20.1 (45.0)	22.8 (51.0)	21.5 (48.0)	23.7 (53.0)	22.8 (51.0)	27.7 (62.0)

Local Climatological Data Annual Summary with Comparative Data for Roswell, New Mexico, National Oceanic and Atmospheric Administration, 2002.

Table 3.2-7 Midland-Odessa Five Year (1987-1991) Annual Joint Frequency Distribution For All Stability Classes Combined
Jan. 1, 1987-Dec. 31, 1991
Wind Speed (mi/hr)
Calm = 2.53 percent
Page 1 of 1

Direction	1-3	4-7	8-12	13-18	19-24	≥ 24.5	Total
N	119	702	722	563	225	57	2388
NNE	71	291	509	556	207	58	1692
NE	64	285	645	776	272	61	2103
ENE	51	382	738	726	170	27	2094
E	69	623	1176	713	95	15	2691
ESE	72	589	1061	557	75	12	2366
SE	70	931	1266	818	134	18	3237
SSE	127	1156	1555	1391	371	48	4648
S	168	1755	2763	3178	820	100	8784
SSW	100	813	1276	807	133	7	3136
SW	61	446	943	757	115	23	2345
WSW	68	356	667	637	191	78	1997
W	84	331	577	517	207	171	1887
WNW	77	244	281	269	75	51	997
NW	91	332	350	224	69	38	1104
NNW	79	500	365	228	80	20	1272
SubTotal	1371	9736	14894	12717	3239	784	42741

Table 3.2-8 Midland-Odessa Five Year (1987-1991) Annual Joint Frequency Distribution Stability Class A
Jan. 1, 1987-Dec. 31, 1991
Wind Speed (mi/hr)
Calm = 0.06 percent
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Direction	1-3	4-7	8-12	13-18	19-24	≥ 24.5	Total
N	3	16	0	0	0	0	19
NNE	3	7	0	0	0	0	10
NE	0	8	0	0	0	0	8
ENE	2	12	0	0	0	0	14
E	3	15	0	0	0	0	18
ESE	3	8	0	0	0	0	11
SE	2	10	0	0	0	0	12
SSE	0	10	0	0	0	0	10
S	3	16	0	0	0	0	19
SSW	2	9	0	0	0	0	11
SW	0	12	0	0	0	0	12
WSW	1	6	0	0	0	0	7
W	0	5	0	0	0	0	5
WNW	0	2	0	0	0	0	2
NW	1	7	0	0	0	0	8
NNW	0	5	0	0	0	0	5
SubTotal	21	145	0	0	0	0	171

Table 3.2-9 Midland-Odessa Five Year (1987-1991) Annual Joint Frequency Distribution Stability Class B
Jan. 1, 1987-Dec. 31, 1991
Wind Speed (mi/hr)
Calm = 0.11 percent
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Direction	1-3	4-7	8-12	13-18	19-24	≥ 24.5	Total
N	20	43	22	0	0	0	85
NNE	17	25	19	0	0	0	61
NE	16	32	22	0	0	0	70
ENE	14	46	36	0	0	0	96
E	6	69	62	0	0	0	137
ESE	17	50	44	0	0	0	111
SE	9	48	45	0	0	0	102
SSE	15	54	64	0	0	0	133
S	25	96	138	0	0	0	259
SSW	12	53	59	0	0	0	124
SW	14	42	49	0	0	0	105
WSW	12	43	43	0	0	0	98
W	16	51	17	0	0	0	84
WNW	11	25	13	0	0	0	49
NW	18	21	14	0	0	0	53
NNW	15	27	9	0	0	0	51
SubTotal	235	722	652	-5	-5	24.5	1618

Table 3.2-10 Midland-Odessa Five Year (1987-1991) Annual Joint Frequency Distribution Stability Class C
Jan. 1, 1987-Dec. 31, 1991
Wind Speed (mi/hr)
Calm = 0.12 percent
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Direction	1-3	4-7	8-12	13-18	19-24	≥ 24.5	Total
N	9	54	124	20	8	3	218
NNE	3	36	87	37	5	1	169
NE	5	37	95	46	11	3	197
ENE	0	52	93	43	4	1	193
E	2	54	164	50	7	0	277
ESE	4	41	147	60	7	0	259
SE	3	36	179	109	10	1	338
SSE	1	65	264	199	52	5	586
S	6	103	527	408	95	19	1158
SSW	5	82	266	124	13	1	491
SW	1	59	238	115	11	2	426
WSW	3	43	180	61	22	7	316
W	5	39	100	76	21	10	251
WNW	4	36	57	25	7	1	130
NW	7	21	51	21	4	0	104
NNW	4	32	48	8	8	3	103
SubTotal	60	787	2616	1397	280	81.5	5216

Table 3.2-11 Midland-Odessa Five Year (1987-1991) Annual Joint Frequency Distribution Stability Class D
Jan. 1, 1987-Dec. 31, 1991
Wind Speed (mi/hr)
Calm = 0.18 percent

Direction	1-3	4-7	8-12	13-18	19-24	≥ 24.5	Total
N	8	112	308	543	217	54	1242
NNE	14	65	302	519	202	57	1159
NE	7	79	389	730	261	58	1524
ENE	6	104	426	683	166	26	1411
E	7	108	550	663	88	15	1431
ESE	13	95	458	497	68	12	1143
SE	5	92	514	709	124	17	1461
SSE	11	98	618	1192	319	43	2281
S	13	151	949	2770	725	81	4689
SSW	3	74	369	683	120	6	1255
SW	1	46	259	642	104	21	1073
WSW	2	42	182	576	169	71	1042
W	4	49	177	441	186	161	1018
WNW	5	29	81	244	68	50	477
NW	3	30	95	203	65	38	434
NNW	7	47	121	220	72	17	484
SubTotal	107	1218	5794	11310	2949	751.5	22124

Table 3.2-12 Midland-Odessa Five Year (1987-1991) Annual Joint Frequency Distribution Stability Class E
Jan. 1, 1987-Dec. 31, 1991
Wind Speed (mi/hr)
Calm = 0.00 percent
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Direction	1-3	4-7	8-12	13-18	19-24	≥ 24.5	Total
N	0	133	268	0	0	0	401
NNE	0	64	101	0	0	0	165
NE	0	66	139	0	0	0	205
ENE	0	81	183	0	0	0	264
E	0	143	400	0	0	0	543
ESE	0	131	412	0	0	0	543
SE	0	236	528	0	0	0	764
SSE	0	259	609	0	0	0	868
S	0	380	1149	0	0	0	1529
SSW	0	145	582	0	0	0	727
SW	0	65	397	0	0	0	462
WSW	0	60	262	0	0	0	322
W	0	42	283	0	0	0	325
WNW	0	36	130	0	0	0	166
NW	0	50	190	0	0	0	240
NNW	0	98	187	0	0	0	285
SubTotal	-2	1986	5816	-5	-5	24.5	7809

Table 3.2-13 Midland-Odessa Five Year (1987-1991) Annual Joint Frequency Distribution Stability Class F
Jan. 1, 1987-Dec. 31, 1991
Wind Speed (mi/hr)
Calm = 2.07 percent
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Direction	1-3	4-7	8-12	13-18	19-24	≥ 24.5	Total
N	79	344	0	0	0	0	423
NNE	34	94	0	0	0	0	128
NE	36	63	0	0	0	0	99
ENE	29	87	0	0	0	0	116
E	51	234	0	0	0	0	285
ESE	35	264	0	0	0	0	299
SE	51	509	0	0	0	0	560
SSE	100	670	0	0	0	0	770
S	121	1009	0	0	0	0	1130
SSW	78	450	0	0	0	0	528
SW	45	222	0	0	0	0	267
WSW	50	162	0	0	0	0	212
W	59	145	0	0	0	0	204
WNW	57	116	0	0	0	0	173
NW	62	203	0	0	0	0	265
NNW	53	291	0	0	0	0	344
SubTotal	938	4860	-4	-5	-5	24.5	5803

Table 3.2-14 Hobbs, New Mexico, Precipitation Data

Precip cm (in)	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sept	Oct	Nov	Dec	Annual
Average	1.3 (0.51)	1.7 (0.66)	1.2 (0.48)	2 (0.78)	6.6 (2.58)	5.2 (2.03)	6.1 (2.42)	6.4 (2.52)	8 (3.13)	3.7 (1.45)	2.2 (0.87)	1.8 (0.72)	46.1 (18.15)
Max	5.2 (2.03)	5.6 (2.21)	7.6 (2.98)	7.3 (2.86)	35.1 (13.83)	13.6 (5.37)	23.9 (9.41)	23 (9.06)	33 (12.99)	20.7 (8.15)	11 (4.33)	12.9 (5.08)	35.1 (13.83)
Min	0.0 (0.0)	0.0 (0.0)	0.0 (0.0)	0.0 (0.0)	0.0 (0.0)	0.0 (0.0)	0.6 (0.22)	0.3 (0.11)	0.2 (0.08)	0.0 (0.0)	0.0 (0.0)	0.0 (0.0)	0.0 (0.0)

Table 3.2-15 Midland-Odessa, Texas, Precipitation Data
1961-1990

Precip cm (in)	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sept	Oct	Nov	Dec	Annual
Average	1.3 (0.53)	1.5 (0.58)	1.1 (0.42)	1.9 (0.73)	4.5 (1.79)	4.3 (1.71)	4.8 (1.89)	4.5 (1.77)	5.9 (2.31)	4.5 (1.77)	1.7 (0.65)	1.7 (0.65)	37.6 (14.8)
Max	9.3 (3.66)	6.5 (2.55)	7.3 (2.86)	7.2 (2.85)	19.4 (7.63)	10.0 (3.93)	21.6 (8.5)	11.3 (4.43)	24.6 (9.7)	18.9 (7.45)	5.9 (2.32)	8.4 (3.3)	24.6 (9.7)
Min	0.0 (0.0)	0.0 (0.0)	T T	0.0 (0.0)	0.1 (0.02)	0.03 (0.01)	T T	0.1 (0.05)	0.0 (0.0)	0.0 (0.0)	0.0 (0.0)	T T	0.0 (0.0)
Max in 24 hours	2.9 (1.15)	3.4 (1.32)	5.6 (2.2)	4.1 (1.62)	12.1 (4.75)	7.8 (3.07)	15.2 (5.99)	6.1 (2.41)	11.1 (4.37)	9.1 (3.59)	5.5 (2.16)	2.3 (0.9)	15.2 (5.99)

T = trace amount

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Table 3.2-16 Roswell, New Mexico, Precipitation Data

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Precip cm (in)	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sept	Oct	Nov	Dec	Annual
Average	1.0 (0.39)	1.0 (0.41)	0.9 (0.35)	1.5 (0.58)	3.3 (1.30)	4.1 (1.62)	5.1 (1.99)	5.9 (2.31)	5.0 (1.98)	3.3 (1.29)	1.3 (0.53)	1.5 (0.59)	33.9 (13.34)
Max	2.6 (1.03)	5.1 (2.02)	7.2 (2.84)	6.3 (2.48)	11.6 (4.57)	12.8 (5.02)	17.5 (6.88)	16.5 (6.48)	16.7 (6.58)	15.0 (5.91)	5.4 (2.11)	7.8 (3.07)	17.5 (6.88)
Min	0.1 (0.03)	0.0 (0.0)	0.0 (0.0)	0.03 (0.01)	T T	0.1 (0.02)	0.03 (0.01)	0.2 (0.07)	0.1 (0.05)	T T	0.0 (0.0)	0.0 (0.0)	0.0 (0.0)
Max in 24 hours	1.7 (0.67)	3.6 (1.41)	5.6 (2.22)	5.7 (2.24)	4.5 (1.77)	7.7 (3.05)	12.5 (4.91)	10.0 (3.94)	6.9 (2.71)	9.9 (3.89)	3.4 (1.33)	2.8 (1.1)	12.5 (4.91)

T = trace amount

Local Climatological Data Annual Summary with Comparative Data for Roswell, New Mexico, National Oceanic and Atmospheric Administration, 2002.

Table 3.2-17 Midland-Odessa, Texas, Snowfall Data
1961-1990

Snowfall cm (in)	**Jan**	**Feb**	**Mar**	**Apr**	**May**	**Jun**	**Jul**	**Aug**	**Sept**	**Oct**	**Nov**	**Dec**	**Annual**
Average	5.6 (2.2)	1.8 (0.7)	0.5 (0.2)	0.3 (0.1)	0.0 (0.0)	0.0 (0.0)	0.0 (0.0)	0.0 (0.0)	0.0 (0.0)	0.* (0.*)	1.3 (0.5)	3.6 (1.4)	13.0 (5.1)
Max	22.9 (9.0)	9.9 (3.9)	15.0 (5.9)	5.1 (2.0)	T T	T T	T T	T T	T T	1.5 (0.6)	20.3 (8.0)	24.9 (9.8)	24.9 (9.8)
Max in 24 hours	17.3 (6.8)	9.9 (3.9)	12.7 (5.0)	5.1 (2.0)	T T	T T	T T	T T	T T	1.5 (0.6)	15.2 (6.0)	24.9 (9.8)	24.9 (9.8)

T = trace amount

0.* indicates the value is between 0.0 and 1.3 cm (0.0 and 0.05 in)

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Table 3.2-18 Roswell, New Mexico, Snowfall Data
1961-1990
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Snowfall cm (in)	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sept	Oct	Nov	Dec	Annual
Average	7.9 (3.1)	6.6 (2.6)	2.3 (0.9)	1.0 (0.4)	0.* (0.*)	0.0 (0.0)	0.0 (0.0)	0.0 (0.0)	0.0 (0.0)	0.8 (0.3)	3.3 (1.3)	8.4 (3.3)	30.2 (11.9)
Max	26.4 (10.4)	42.9 (16.9)	12.2 (4.8)	13.5 (5.3)	2.0 (0.8)	2.5 (1.0)	0.0 (0.0)	0.0 (0.0)	2.5 (1.0)	10.7 (4.2)	31.2 (12.3)	53.3 (21.0)	53.3 (21.0)
Max in 24 hours	18.5 (7.3)	41.9 (16.5)	12.2 (4.8)	10.2 (4.0)	5.1 (2.0)	2.5 (1.0)	0.0 (0.0)	0.0 (0.0)	2.5 (1.0)	7.9 (3.1)	16.0 (6.3)	24.6 (9.7)	41.9 (16.5)

0.* indicates the value is between 0.0 and 1.3 cm (0.0 and 0.05 in)

Local Climatological Data Annual Summary with Comparative Data for Roswell, New Mexico, National Oceanic and Atmospheric Administration, 2002.

Table 3.2-19 Straight Wind Hazard Assessment
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Annual Probability	Expected Wind Speed km/hr (mi/hr)	Upper Bound Wind Speed km/hr (mi/hr)	Lower Bound Wind Speed km/hr (mi/hr)
1E-01	134 (83)	146 (91)	119 (74)
1E-02	162 (101)	188 (117)	138 (86)
1E-03	193 (120)	230 (143)	156 (97)
1E-04	222 (138)	271(169)	174 (108)
1E-05	252 (157)	312 (194)	191 (119)
1E-06	282 (175)	354 (220)	209 (130)

Table 3.2-20 Location of Recorded Earthquakes Within a 322 km (200 mi) Radius of the NEF Site

NEF Site Coordinates	Longitude	Latitude
	-103.0820	32.4360

Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth[1] (km)	Focal Depth[1] (mi)	MAG[2]	MAG Type[3]	Epicentral Distance (km)	Epicentral Distance (mi)	Data Sources[4]
1931	8	16	-104.60	30.70			6.00	M	240.3	149.3	UTIG
1949	5	23	-105.20	34.60			4.50	M	310.0	192.6	NMTH
1955	1	27	-104.50	30.60			3.30	M	244.0	151.6	UTIG
1962	3	6	-104.80	31.20			3.50	M	212.3	131.9	UTIG
1963	12	19	-104.27	34.82			3.40	M	287.0	178.3	NMTR
1964	2	11	-103.94	34.23			2.10	M	214.2	133.1	NMTR
1964	3	3	-103.60	34.84			2.90	M	271.0	168.4	NMTR
1964	6	19	-105.77	32.95			1.90	M	257.4	159.9	NMTR
1964	8	14	-102.94	31.97			1.90	M	53.1	33.0	NMTR
1964	9	7	-102.92	31.94			1.60	M	56.9	35.3	NMTR
1964	11	8	-103.10	31.90			3.00	M	59.5	37.0	UTIG
1964	11	21	-103.10	31.90			3.10	M	59.5	37.0	UTIG
1964	11	27	-102.97	31.89			1.90	M	61.1	38.0	NMTR
1965	1	21	-102.85	32.02			1.30	M	50.9	31.6	NMTR
1965	2	3	-103.10	31.90			3.30	M	59.5	37.0	UTIG
1965	8	30	-103.00	31.90			3.50	M	60.0	37.3	UTIG
1966	8	14	-103.00	31.90			3.40	M	60.0	37.3	UTIG
1966	9	17	-103.98	34.89			2.70	M	284.6	176.9	NMTR
1966	10	6	-104.12	35.13			2.90	M	314.4	195.4	NMTR
1966	11	26	-105.44	30.95			3.50	M	277.5	172.4	NMTR
1968	3	23	-105.91	32.67			2.60	M	265.7	165.1	NMTR
1968	5	2	-105.24	33.10			2.60	M	214.3	133.1	NMTR
1969	6	1	-105.21	34.20			1.90	M	277.7	172.5	NMTR
1969	6	8	-105.19	34.15			2.60	M	272.8	169.5	NMTR
1971	7	30	-103.00	31.72	10.0	6.2	3.00	mb	79.9	49.6	ANSS
1971	7	31	-103.06	31.70	10.0	6.2	3.40	mb	81.4	50.6	ANSS
1971	9	24	-103.20	31.60			3.20	M	93.5	58.1	UTIG
1972	7	26	-104.01	32.57			3.10	M	88.3	54.9	NMTR
1973	3	17	-102.36	31.59			2.50	M	115.7	71.9	NMTR
1973	8	2	-105.56	31.04			3.60	M	280.7	174.5	NMTR
1973	8	4	-103.22	35.11			3.00	M	296.6	184.3	NMTR
1974	7	31	-104.19	33.11			0.00	M	128.0	79.5	NMTR
1974	10	2	-100.86	31.87			0.00	M	217.7	135.3	NMTR
1974	10	27	-104.83	30.63			0.00	M	259.6	161.3	NMTR
1974	11	12	-102.67	32.14			0.00	M	51.0	31.7	NMTR
1974	11	21	-102.75	32.07			0.00	M	51.0	31.7	NMTR

Table 3.2-20 Location of Recorded Earthquakes Within a 322 km (200 mi) Radius of the NEF Site

Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth[1] (km)	Focal Depth[1] (mi)	MAG[2]	MAG Type[3]	Epicentral Distance (km)	Epicentral Distance (mi)	Data Sources[4]
1974	11	22	-101.26	32.94			0.00	M	179.2	111.3	NMTR
1974	11	22	-105.21	33.78			0.00	M	247.7	153.9	NMTR
1974	11	28	-103.94	32.58			0.00	M	82.2	51.1	NMTR
1974	11	28	-104.14	32.31	5.0	3.1	3.90	mb	100.4	62.4	ANSS
1974	12	30	-103.10	30.90			3.70	M	170.5	106.0	UTIG
1975	1	30	-103.08	30.95			2.10	M	165.1	102.6	NMTR
1975	2	2	-103.19	35.05			3.00	M	290.7	180.6	NMTR
1975	4	8	-101.69	32.18			0.00	M	133.9	83.2	NMTR
1975	7	25	-102.62	29.82			0.00	M	293.4	182.3	NMTR
1975	8	1	-104.60	30.49			0.00	M	259.5	161.3	NMTR
1975	8	1	-104.00	31.40			3.00	M	143.9	89.4	UTIG
1975	8	3	-104.45	30.71			0.00	M	231.0	143.5	NMTR
1975	10	10	-105.02	33.36			0.00	M	207.4	128.9	NMTR
1975	12	12	-102.31	31.61			3.00	M	117.5	73.0	NMTR
1976	1	10	-102.76	31.79			0.00	M	78.4	48.7	NMTR
1976	1	15	-102.32	30.98			0.00	M	176.6	109.7	NMTR
1976	1	19	-103.09	31.90			3.50	M	59.5	37.0	UTIG
1976	1	21	-102.29	30.95			0.00	M	180.8	112.4	NMTR
1976	1	22	-103.07	31.90	1.0	0.6	2.80	un	59.5	37.0	ANSS
1976	1	25	-103.08	31.90	2.0	1.2	3.90	un	59.3	36.8	ANSS
1976	1	28	-100.89	31.99			0.00	M	211.8	131.6	NMTR
1976	2	4	-103.53	31.68			0.00	M	94.1	58.4	NMTR
1976	2	14	-102.47	31.63			0.00	M	106.2	66.0	NMTR
1976	3	5	-102.25	31.66			0.00	M	116.7	72.5	NMTR
1976	3	15	-102.58	32.50			0.00	M	47.3	29.4	NMTR
1976	3	18	-102.96	32.33			0.00	M	16.5	10.3	NMTR
1976	3	20	-104.94	31.27			0.00	M	217.4	135.1	NMTR
1976	3	20	-103.06	32.22			0.00	M	24.4	15.2	NMTR
1976	3	27	-103.07	32.22			0.00	M	23.7	14.7	NMTR
1976	4	3	-103.10	31.24			0.00	M	132.5	82.3	NMTR
1976	4	12	-103.00	32.27			0.00	M	20.2	12.5	NMTR
1976	4	21	-102.89	32.25			0.00	M	27.7	17.2	NMTR
1976	4	30	-103.09	31.98			0.00	M	50.7	31.5	NMTR
1976	4	30	-103.11	31.92			0.00	M	57.6	35.8	NMTR
1976	5	1	-103.06	32.37			0.00	M	8.0	5.0	NMTR
1976	5	3	-105.66	32.41			0.00	M	241.7	150.2	NMTR
1976	5	3	-103.20	32.03			0.00	M	47.0	29.2	NMTR
1976	5	3	-103.03	32.03			0.00	M	45.6	28.3	NMTR
1976	5	4	-103.23	31.86			0.00	M	65.3	40.6	NMTR
1976	5	6	-103.18	31.97			0.00	M	53.1	33.0	NMTR
1976	5	6	-103.16	31.87			0.00	M	63.3	39.3	NMTR
1976	5	11	-102.92	32.29			0.00	M	22.2	13.8	NMTR
1976	5	21	-105.59	32.49			0.00	M	234.9	146.0	NMTR
1976	6	14	-102.49	31.52			0.00	M	116.5	72.4	NMTR
1976	6	15	-102.34	31.56			0.00	M	120.0	74.6	NMTR

Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth[1] (km)	Focal Depth[1] (mi)	MAG[2]	MAG Type[3]	Epicentral Distance (km)	Epicentral Distance (mi)	Data Sources[4]
1976	6	15	-102.37	31.60			0.00	M	115.0	71.5	NMTR
1976	7	28	-102.29	33.02			0.00	M	98.7	61.4	NMTR
1976	8	5	-101.73	30.87			0.00	M	216.3	134.4	NMTR
1976	8	5	-103.00	31.60			3.00	M	93.1	57.9	UTIG
1976	8	6	-102.59	31.78			2.10	M	86.3	53.6	NMTR
1976	8	10	-102.03	31.77			0.00	M	123.8	76.9	NMTR
1976	8	10	-102.06	31.79			0.00	M	119.5	74.3	NMTR
1976	8	25	-101.94	31.55			0.00	M	146.1	90.8	NMTR
1976	8	26	-102.01	31.84			0.00	M	120.8	75.1	NMTR
1976	8	30	-101.98	31.57			0.00	M	141.7	88.0	NMTR
1976	8	31	-102.18	31.46			0.00	M	137.4	85.4	NMTR
1976	9	3	-103.48	31.55			2.00	M	105.2	65.4	NMTR
1976	9	5	-102.74	32.23			0.00	M	39.3	24.4	NMTR
1976	9	17	-103.06	32.24			0.00	M	22.4	13.9	NMTR
1976	9	17	-102.50	31.40			3.10	M	127.4	79.2	UTIG
1976	9	19	-104.57	30.47			0.00	M	259.7	161.4	NMTR
1976	10	22	-102.16	31.55			0.00	M	131.6	81.8	NMTR
1976	10	23	-102.38	31.62			0.00	M	112.2	69.7	NMTR
1976	10	25	-102.53	31.84			0.00	M	84.3	52.4	NMTR
1976	10	26	-103.28	31.33			2.40	M	124.2	77.2	NMTR
1976	11	3	-102.27	30.92			0.00	M	185.6	115.3	NMTR
1976	12	12	-102.46	31.57			2.80	M	112.5	69.9	NMTR
1976	12	12	-102.49	31.61			1.90	M	107.3	66.6	NMTR
1976	12	15	-102.22	31.59			1.40	M	124.2	77.2	NMTR
1976	12	18	-103.02	31.62			1.80	M	90.8	56.4	NMTR
1976	12	19	-102.45	31.87			2.20	M	86.0	53.5	NMTR
1976	12	19	-103.14	32.25			1.80	M	20.9	13.0	NMTR
1976	12	19	-103.08	32.27			2.70	M	18.7	11.6	NMTR
1977	1	29	-104.59	30.58			0.00	M	250.3	155.5	NMTR
1977	2	4	-104.70	30.59			0.00	M	256.1	159.2	NMTR
1977	2	18	-103.05	32.24			0.00	M	21.7	13.5	NMTR
1977	3	5	-102.66	31.16			0.00	M	146.9	91.3	NMTR
1977	3	14	-101.01	33.04			0.00	M	204.7	127.2	NMTR
1977	3	20	-103.10	32.21			0.00	M	25.5	15.8	NMTR
1977	3	29	-103.28	31.60			0.00	M	94.2	58.5	NMTR
1977	4	3	-103.17	31.49			1.90	M	105.3	65.5	NMTR
1977	4	3	-103.20	31.47			0.00	M	107.8	67.0	NMTR
1977	4	4	-103.36	31.00			0.00	M	161.4	100.3	NMTR
1977	4	7	-103.05	32.19			0.00	M	27.7	17.2	NMTR
1977	4	7	-102.70	31.32			0.00	M	129.3	80.3	NMTR
1977	4	7	-102.94	31.35			0.00	M	120.9	75.1	NMTR
1977	4	12	-102.55	31.28			0.00	M	137.4	85.4	NMTR
1977	4	17	-102.35	31.50			0.00	M	124.7	77.5	NMTR
1977	4	18	-103.25	31.60			0.00	M	93.7	58.2	NMTR
1977	4	22	-103.02	32.18			0.00	M	28.8	17.9	NMTR
1977	4	25	-102.81	32.07			0.00	M	47.9	29.8	NMTR

Year	Month	Day	Longitude	Latitude	Focal Depth[1]		MAG[2]	MAG Type[3]	Epicentral Distance		Data Sources[4]
			(°W)	(°N)	(km)	(mi)			(km)	(mi)	
1977	4	26	-103.08	31.90	4.0	2.5	3.30	un	59.3	36.8	ANSS
1977	4	28	-102.52	31.83			0.00	M	86.1	53.5	NMTR
1977	4	28	-101.99	31.87			0.00	M	120.6	75.0	NMTR
1977	4	29	-102.65	31.77			0.00	M	84.0	52.2	NMTR
1977	6	7	-100.75	33.06	5.0	3.1	4.00	un	228.5	142.0	ANSS
1977	6	8	-100.83	32.83			0.00	M	215.4	133.9	NMTR
1977	6	8	-100.82	32.92			0.00	M	218.4	135.7	NMTR
1977	6	8	-101.04	32.87			0.00	M	196.4	122.1	NMTR
1977	6	17	-100.95	32.90			2.70	M	206.1	128.1	NMTR
1977	6	28	-103.30	31.54			2.30	M	101.6	63.1	NMTR
1977	7	1	-103.34	31.50			2.00	M	106.7	66.3	NMTR
1977	7	11	-102.62	31.80			0.00	M	83.1	51.6	NMTR
1977	7	11	-102.68	31.79			0.00	M	81.4	50.6	NMTR
1977	7	12	-102.64	31.77			0.00	M	84.6	52.6	NMTR
1977	7	18	-102.70	31.78			0.00	M	81.4	50.6	NMTR
1977	7	22	-102.72	31.80			0.00	M	78.2	48.6	NMTR
1977	7	22	-102.70	31.80			3.00	M	79.2	49.2	UTIG
1977	7	24	-102.70	31.79			0.00	M	79.7	49.5	NMTR
1977	8	20	-103.33	31.60			1.90	M	95.7	59.5	NMTR
1977	8	21	-104.91	30.54			0.00	M	272.4	169.3	NMTR
1977	10	13	-100.81	32.91			2.20	M	218.8	135.9	NMTR
1977	10	17	-102.46	31.57			1.80	M	112.6	69.9	NMTR
1977	11	14	-104.96	31.52			0.00	M	203.7	126.6	NMTR
1977	11	27	-101.14	33.02			0.00	M	192.7	119.8	NMTR
1977	11	28	-100.84	32.95	5.0	3.1	3.50	un	217.4	135.1	ANSS
1977	12	16	-102.40	31.52			0.00	M	120.2	74.7	NMTR
1977	12	21	-102.41	31.52			0.00	M	120.3	74.7	NMTR
1977	12	31	-102.46	31.60			2.10	M	109.7	68.2	NMTR
1978	1	2	-102.53	31.60			2.20	M	106.3	66.1	NMTR
1978	1	12	-102.30	31.49			0.00	M	128.1	79.6	NMTR
1978	1	15	-101.70	31.36			0.00	M	177.0	110.0	NMTR
1978	1	18	-103.23	31.61			0.00	M	92.9	57.7	NMTR
1978	1	19	-103.71	32.56			0.00	M	60.5	37.6	NMTR
1978	2	5	-102.60	31.89			0.00	M	76.2	47.4	NMTR
1978	2	5	-104.55	31.41			0.00	M	179.5	111.5	NMTR
1978	2	18	-104.69	31.21			2.30	M	203.8	126.6	NMTR
1978	3	2	-103.06	32.82			1.50	M	42.5	26.4	NMTR
1978	3	2	-102.38	31.58			3.30	M	115.4	71.7	NMTR
1978	3	2	-102.61	31.59			2.10	M	103.9	64.6	NMTR
1978	3	2	-102.56	31.55			3.50	M	109.9	68.3	UTIG
1978	3	19	-102.49	31.47			1.60	M	120.5	74.9	NMTR
1978	6	16	-100.80	33.00			3.40	M	222.1	138.0	UTIG
1978	6	16	-100.77	33.03	10.0	6.2	5.30	un	226.1	140.5	ANSS
1978	6	29	-102.42	31.08			3.20	M	163.1	101.4	NMTR
1978	7	5	-102.20	31.61			0.00	M	123.2	76.5	NMTR
1978	7	18	-104.36	30.36			0.00	M	260.4	161.8	NMTR

Year	Month	Day	Longitude	Latitude	Focal Depth[1]		MAG[2]	MAG Type[3]	Epicentral Distance		Data Sources[4]
			(°W)	(°N)	(km)	(mi)			(km)	(mi)	
1978	7	21	-102.77	31.34			0.00	M	125.0	77.7	NMTR
1978	8	14	-102.18	31.58			2.20	M	127.4	79.2	NMTR
1978	9	29	-102.42	31.52			0.00	M	119.2	74.1	NMTR
1978	9	30	-102.17	31.36			0.00	M	146.7	91.1	NMTR
1978	10	2	-102.43	31.53			0.00	M	117.6	73.1	NMTR
1978	10	2	-102.19	31.51			0.00	M	132.5	82.3	NMTR
1978	10	2	-102.36	31.48			0.00	M	126.4	78.5	NMTR
1978	10	3	-102.99	31.90			0.00	M	59.7	37.1	NMTR
1978	10	6	-102.36	31.55			0.00	M	119.8	74.4	NMTR
1979	4	28	-104.72	30.47			0.00	M	267.7	166.3	NMTR
1979	7	17	-103.73	32.65			2.00	M	65.4	40.6	NMTR
1979	8	3	-100.81	32.87			2.40	M	217.5	135.1	NMTR
1980	1	21	-105.00	34.20			1.30	M	264.2	164.2	NMTR
1980	3	21	-102.34	31.57			1.60	M	118.5	73.6	NMTR
1981	8	13	-102.70	31.90			2.20	M	69.7	43.3	NMTR
1981	9	16	-105.23	33.72			1.80	M	245.2	152.4	NMTR
1982	1	4	-102.49	31.18	5.0	3.1	3.90	un	149.9	93.2	ANSS
1982	4	26	-100.84	33.02	5.0	3.1	2.80	un	218.8	136.0	ANSS
1982	5	1	-103.04	32.33			2.10	M	12.3	7.6	NMTR
1982	10	17	-102.71	30.90			2.00	M	174.0	108.1	NMTR
1982	10	26	-103.59	33.67			1.50	M	144.6	89.8	NMTR
1982	10	26	-103.61	33.63			1.50	M	141.3	87.8	NMTR
1982	11	25	-100.78	32.89			2.30	M	220.7	137.1	NMTR
1982	11	28	-100.84	33.00	5.0	3.1	3.30	un	218.4	135.7	ANSS
1983	1	9	-104.19	30.65			1.90	M	224.3	139.4	NMTR
1983	1	12	-105.19	34.32			1.50	M	286.7	178.2	NMTR
1983	1	29	-102.08	31.75			2.20	M	121.2	75.3	NMTR
1983	3	3	-104.35	29.96			2.80	M	299.6	186.2	NMTR
1983	6	5	-105.35	32.52			1.30	M	212.6	132.1	NMTR
1983	6	21	-103.58	33.63			1.60	M	140.9	87.5	NMTR
1983	7	21	-105.14	30.97			1.60	M	253.4	157.5	NMTR
1983	8	4	-105.14	32.57			1.30	M	193.4	120.2	NMTR
1983	8	19	-102.23	31.31			1.80	M	148.8	92.5	NMTR
1983	8	22	-105.08	34.06			1.30	M	258.6	160.7	NMTR
1983	8	23	-105.52	31.17			2.10	M	269.7	167.6	NMTR
1983	8	26	-102.53	33.62			1.60	M	140.9	87.5	NMTR
1983	8	29	-100.62	31.80			2.60	M	242.0	150.4	NMTR
1983	9	15	-104.43	34.92			3.10	M	302.6	188.1	NMTR
1983	9	29	-104.45	34.89			2.70	M	300.0	186.4	NMTR
1983	9	30	-103.97	30.57			1.70	M	224.0	139.2	NMTR
1983	12	1	-101.99	31.86			1.40	M	121.1	75.3	NMTR
1983	12	3	-103.32	30.97			2.10	M	164.1	102.0	NMTR
1983	12	26	-102.88	30.77			1.70	M	186.4	115.8	NMTR
1984	1	2	-102.12	31.81			1.80	M	114.4	71.1	NMTR
1984	1	3	-102.69	31.21			1.70	M	141.3	87.8	NMTR
1984	1	3	-103.04	30.76			2.00	M	186.3	115.8	NMTR

Table 3.2-20 Location of Recorded Earthquakes Within a 322 km (200 mi) Radius of the NEF Site

Year	Month	Day	Longitude	Latitude	Focal Depth[1]		MAG[2]	MAG Type[3]	Epicentral Distance		Data Sources[4]
			(°W)	(°N)	(km)	(mi)			(km)	(mi)	
1984	1	16	-102.20	31.56			1.40	M	127.5	79.2	NMTR
1984	3	2	-104.84	30.81			1.90	M	245.5	152.5	NMTR
1984	3	23	-100.78	32.45			1.50	M	215.2	133.7	NMTR
1984	5	21	-102.59	31.14			1.30	M	151.3	94.0	NMTR
1984	5	21	-102.23	35.07	5.0	3.1	3.10	un	302.5	188.0	ANSS
1984	6	27	-102.48	31.22			2.00	M	146.5	91.0	NMTR
1984	7	17	-105.77	32.85			1.30	M	255.7	158.9	NMTR
1984	8	18	-103.56	30.78			1.80	M	189.8	118.0	NMTR
1984	8	24	-104.48	30.67			1.30	M	236.8	147.1	NMTR
1984	8	26	-104.27	30.38			2.10	M	254.4	158.1	NMTR
1984	9	11	-100.70	31.99	5.0	3.1	3.20	un	229.4	142.5	ANSS
1984	9	19	-100.69	32.03	5.0	3.1	3.00	un	229.3	142.5	ANSS
1984	9	27	-103.42	32.59			1.60	M	36.0	22.4	NMTR
1984	10	4	-102.70	33.58			1.30	M	132.3	82.2	NMTR
1984	10	4	-102.24	31.65			1.30	M	118.4	73.6	NMTR
1984	10	11	-100.56	31.95			2.40	M	243.2	151.1	NMTR
1984	10	27	-104.56	30.62			1.70	M	245.1	152.3	NMTR
1984	11	27	-105.41	33.57			1.60	M	250.6	155.7	NMTR
1984	12	4	-101.93	30.10			2.30	M	281.6	175.0	NMTR
1984	12	4	-103.21	32.64			2.10	M	25.4	15.8	NMTR
1984	12	4	-103.56	32.27	5.0	3.1	2.90	un	48.3	30.0	ANSS
1984	12	12	-105.61	33.36			1.50	M	256.9	159.6	NMTR
1985	2	21	-100.75	32.88			1.40	M	223.3	138.7	NMTR
1985	2	21	-100.81	32.72			1.50	M	214.6	133.4	NMTR
1985	3	9	-105.12	33.97			1.30	M	254.4	158.1	NMTR
1985	5	3	-104.95	31.04			1.90	M	234.5	145.7	NMTR
1985	6	1	-102.83	31.06			1.50	M	154.6	96.0	NMTR
1985	6	2	-102.28	31.18			1.60	M	158.7	98.6	NMTR
1985	6	12	-103.90	34.64			1.60	M	255.9	159.0	NMTR
1985	8	2	-104.34	32.48			1.40	M	118.0	73.3	NMTR
1985	9	5	-103.77	33.66			1.80	M	150.1	93.3	NMTR
1985	9	18	-103.42	30.90			2.00	M	173.1	107.6	NMTR
1985	10	21	-101.88	32.04			1.30	M	121.3	75.4	NMTR
1985	11	13	-103.08	32.10			1.80	M	37.8	23.5	NMTR
1985	11	28	-101.99	31.61			1.80	M	138.2	85.9	NMTR
1985	12	5	-102.94	32.42			1.60	M	13.9	8.6	NMTR
1986	1	25	-100.73	32.06	5.0	3.1	2.90	un	224.3	139.4	ANSS
1986	1	30	-104.01	33.54			1.90	M	150.1	93.3	NMTR
1986	1	30	-100.69	32.07	5.0	3.1	3.30	un	228.0	141.7	ANSS
1986	2	7	-105.44	32.54			1.40	M	221.0	137.3	NMTR
1986	2	14	-100.76	31.53			2.60	M	240.9	149.7	NMTR
1986	3	1	-102.57	31.16			1.70	M	149.6	92.9	NMTR
1986	3	11	-105.08	32.11			2.00	M	190.7	118.5	NMTR
1986	3	21	-105.64	33.43			1.60	M	262.8	163.3	NMTR
1986	5	28	-105.12	31.76			1.60	M	205.8	127.9	NMTR
1986	6	12	-102.22	31.77			1.80	M	109.6	68.1	NMTR

Year	Month	Day	Longitude	Latitude	Focal Depth[1]		MAG[2]	MAG Type[3]	Epicentral Distance		Data Sources[4]
			(°W)	(°N)	(km)	(mi)			(km)	(mi)	
1986	6	27	-102.01	32.06			2.20	M	109.3	67.9	NMTR
1986	7	9	-102.48	31.55			1.60	M	113.3	70.4	NMTR
1986	7	20	-105.00	33.47			1.50	M	212.8	132.2	NMTR
1986	8	2	-103.79	33.68			1.70	M	153.4	95.3	NMTR
1986	8	6	-103.03	33.86			2.40	M	158.4	98.5	NMTR
1986	8	14	-104.66	32.53			1.30	M	148.0	92.0	NMTR
1986	8	15	-103.43	33.14			1.70	M	84.2	52.3	NMTR
1986	8	29	-102.41	31.31			1.40	M	140.1	87.1	NMTR
1986	9	18	-102.37	31.51			1.80	M	123.2	76.5	NMTR
1986	10	18	-102.69	30.07			1.60	M	265.4	164.9	NMTR
1986	10	25	-102.13	31.60			1.70	M	129.0	80.2	NMTR
1986	11	3	-104.64	31.09			2.00	M	209.5	130.2	NMTR
1986	11	6	-104.58	32.55			1.60	M	140.4	87.2	NMTR
1986	11	17	-100.73	33.08			2.00	M	230.6	143.3	NMTR
1986	11	24	-102.16	31.68			2.00	M	121.1	75.3	NMTR
1986	12	6	-102.16	31.59			2.40	M	127.6	79.3	NMTR
1986	12	6	-102.23	31.47			2.10	M	133.9	83.2	NMTR
1986	12	6	-102.17	31.65			1.70	M	122.0	75.8	NMTR
1986	12	6	-102.09	31.72			2.20	M	122.6	76.2	NMTR
1986	12	15	-103.19	35.07			1.50	M	292.9	182.0	NMTR
1986	12	15	-102.02	31.76			1.50	M	125.0	77.7	NMTR
1987	1	25	-104.86	31.74			1.70	M	184.3	114.5	NMTR
1987	2	9	-103.45	30.69			2.30	M	196.8	122.3	NMTR
1987	2	9	-101.96	31.86			1.60	M	123.6	76.8	NMTR
1987	2	12	-101.94	31.66			1.60	M	137.9	85.7	NMTR
1987	2	17	-104.52	30.60			2.10	M	244.8	152.1	NMTR
1987	3	2	-105.08	30.78			1.80	M	263.6	163.8	NMTR
1987	3	3	-105.44	31.17			1.50	M	263.4	163.7	NMTR
1987	3	10	-105.66	31.13			1.50	M	282.7	175.7	NMTR
1987	3	26	-103.28	30.96			2.60	M	165.2	102.6	NMTR
1987	3	31	-104.95	31.52			2.80	M	203.4	126.4	NMTR
1987	4	23	-105.02	32.03			1.60	M	187.7	116.7	NMTR
1987	4	25	-105.22	33.97			1.90	M	261.2	162.3	NMTR
1987	4	29	-105.92	32.67			2.30	M	267.0	165.9	NMTR
1987	7	5	-104.77	30.85			2.00	M	237.5	147.6	NMTR
1987	7	23	-103.03	35.29			1.90	M	316.9	196.9	NMTR
1987	7	30	-103.87	34.54			1.50	M	244.4	151.9	NMTR
1987	8	4	-102.12	31.87			1.70	M	110.1	68.4	NMTR
1987	9	11	-103.62	33.61			2.00	M	139.1	86.4	NMTR
1987	9	21	-103.74	33.68			1.80	M	150.6	93.6	NMTR
1987	10	1	-105.16	30.47			1.60	M	294.1	182.7	NMTR
1987	10	1	-103.76	33.66			1.50	M	150.0	93.2	NMTR
1987	10	9	-104.59	31.07			1.40	M	208.4	129.5	NMTR
1987	10	31	-105.31	32.86			1.30	M	213.8	132.9	NMTR
1987	11	3	-103.71	33.70			1.30	M	151.6	94.2	NMTR
1987	11	17	-101.97	32.06			1.60	M	112.9	70.1	NMTR

Table 3.2-20 Location of Recorded Earthquakes Within a 322 km (200 mi) Radius of the NEF Site

Year	Month	Day	Longitude	Latitude	Focal Depth[1]		MAG[2]	MAG Type[3]	Epicentral Distance		Data Sources[4]
			(°W)	(°N)	(km)	(mi)			(km)	(mi)	
1987	12	6	-102.76	31.83			1.60	M	74.2	46.1	NMTR
1987	12	20	-103.07	32.29			2.20	M	15.8	9.8	NMTR
1987	12	28	-102.25	31.47			2.10	M	133.3	82.8	NMTR
1987	12	29	-102.11	31.58			1.50	M	132.1	82.1	NMTR
1988	1	26	-102.42	31.24			2.30	M	146.4	90.9	NMTR
1988	2	14	-102.06	31.78			1.40	M	121.0	75.2	NMTR
1988	2	21	-103.02	30.45			1.40	M	220.3	136.9	NMTR
1988	2	27	-103.75	33.67			1.80	M	150.3	93.4	NMTR
1988	3	9	-102.44	31.24			1.70	M	146.0	90.7	NMTR
1988	3	15	-105.52	31.72			1.30	M	242.7	150.8	NMTR
1988	3	17	-102.20	31.66			1.60	M	119.8	74.4	NMTR
1988	4	5	-102.33	31.44			2.10	M	131.6	81.8	NMTR
1988	4	6	-102.09	31.94			1.30	M	107.9	67.1	NMTR
1988	5	3	-104.39	30.52			1.30	M	246.2	153.0	NMTR
1988	5	10	-105.20	30.96			1.40	M	258.4	160.6	NMTR
1988	5	27	-102.12	31.78			1.30	M	116.1	72.1	NMTR
1988	5	27	-102.02	32.06			1.30	M	108.3	67.3	NMTR
1988	7	4	-100.74	33.74			2.00	M	261.5	162.5	NMTR
1988	7	11	-103.25	35.28			1.90	M	316.6	196.7	NMTR
1988	7	20	-102.43	29.77			2.20	M	301.9	187.6	NMTR
1988	7	25	-104.91	31.98			1.50	M	178.9	111.2	NMTR
1988	7	26	-105.14	30.94			1.50	M	255.5	158.8	NMTR
1988	8	23	-102.02	32.26			1.50	M	101.1	62.8	NMTR
1988	9	15	-103.32	31.68			1.50	M	86.7	53.9	NMTR
1988	9	19	-102.45	32.46			2.00	M	59.3	36.8	NMTR
1988	10	2	-103.79	33.63			1.30	M	147.8	91.8	NMTR
1988	11	10	-102.40	31.55			1.90	M	117.3	72.9	NMTR
1989	1	9	-102.59	31.44			1.80	M	119.6	74.3	NMTR
1989	1	9	-102.12	31.78			1.30	M	116.5	72.4	NMTR
1989	1	20	-101.97	32.08			1.90	M	112.1	69.6	NMTR
1989	2	21	-103.39	35.29			2.30	M	318.4	197.8	NMTR
1989	3	19	-103.55	31.19			1.50	M	145.2	90.2	NMTR
1989	3	21	-102.33	31.42			1.50	M	133.5	83.0	NMTR
1989	3	30	-102.86	33.24			1.40	M	91.5	56.9	NMTR
1989	6	5	-102.09	32.10			2.10	M	100.1	62.2	NMTR
1989	6	23	-102.23	31.59			1.60	M	123.2	76.6	NMTR
1989	6	28	-105.08	30.93			2.30	M	252.3	156.8	NMTR
1989	7	13	-105.27	33.53			1.50	M	237.1	147.3	NMTR
1989	7	24	-100.93	32.92			1.60	M	208.3	129.5	NMTR
1989	7	25	-101.76	30.90			2.10	M	211.2	131.3	NMTR
1989	8	8	-102.70	31.30			2.30	M	131.3	81.6	NMTR
1989	8	16	-101.96	31.70			1.60	M	133.3	82.8	NMTR
1989	9	5	-102.50	34.25			2.50	M	208.9	129.8	NMTR
1989	11	2	-100.94	33.02			2.00	M	210.4	130.7	NMTR
1989	11	16	-103.12	35.11			2.60	M	296.7	184.4	NMTR
1989	12	7	-103.67	34.58			1.40	M	244.1	151.7	NMTR

Year	Month	Day	Longitude	Latitude	Focal Depth[1]		MAG[2]	MAG Type[3]	Epicentral Distance		Data Sources[4]
			(°W)	(°N)	(km)	(mi)			(km)	(mi)	
1989	12	28	-101.06	31.70			2.10	M	207.6	129.0	NMTR
1989	12	28	-100.96	32.04			1.70	M	203.9	126.7	NMTR
1990	1	16	-105.32	31.74			1.80	M	224.4	139.4	NMTR
1990	3	4	-103.92	30.53			1.70	M	226.3	140.6	NMTR
1990	3	30	-100.53	32.96			2.30	M	245.1	152.3	NMTR
1990	3	30	-100.56	32.99			2.20	M	243.5	151.3	NMTR
1990	4	6	-103.36	31.51			1.90	M	106.3	66.0	NMTR
1990	5	10	-102.37	31.14			2.20	M	159.2	98.9	NMTR
1990	5	10	-101.96	32.13			1.60	M	110.9	68.9	NMTR
1990	5	16	-102.04	31.86			2.40	M	117.2	72.8	NMTR
1990	5	22	-102.09	30.24			2.20	M	261.5	162.5	NMTR
1990	6	22	-100.76	32.58			2.20	M	218.3	135.7	NMTR
1990	7	3	-102.22	31.44			1.50	M	137.6	85.5	NMTR
1990	7	13	-101.81	34.86			2.70	M	293.9	182.6	NMTR
1990	8	3	-100.69	32.21			3.40	M	225.6	140.2	NMTR
1990	8	9	-102.67	31.21			1.90	M	141.8	88.1	NMTR
1990	8	14	-102.26	31.39			1.80	M	139.8	86.9	NMTR
1990	8	25	-102.01	31.91			1.80	M	116.0	72.1	NMTR
1990	10	8	-105.12	30.94			1.30	M	254.0	157.8	NMTR
1990	12	20	-103.14	35.27			2.50	M	315.1	195.8	NMTR
1991	1	1	-105.27	32.44			1.60	M	205.4	127.6	NMTR
1991	1	29	-103.04	32.89			1.40	M	50.8	31.6	NMTR
1991	2	3	-104.49	32.81			1.30	M	137.7	85.6	NMTR
1991	2	3	-103.96	35.00			2.10	M	296.2	184.0	NMTR
1991	3	10	-103.97	30.47			2.10	M	234.3	145.6	NMTR
1991	3	10	-103.33	33.58			2.00	M	128.8	80.0	NMTR
1991	4	8	-103.13	34.98			2.10	M	282.4	175.5	NMTR
1991	5	16	-103.75	33.67			2.00	M	150.4	93.5	NMTR
1991	6	4	-102.31	32.05			2.00	M	83.9	52.1	NMTR
1991	7	16	-101.12	33.09			2.10	M	197.3	122.6	NMTR
1991	8	1	-104.02	34.59			2.70	M	254.6	158.2	NMTR
1991	8	7	-104.81	31.62			1.80	M	186.1	115.6	NMTR
1991	8	17	-100.99	32.09			2.00	M	200.2	124.4	NMTR
1991	9	22	-101.30	31.32			2.10	M	209.2	130.0	NMTR
1991	9	28	-103.77	33.63			1.70	M	147.3	91.6	NMTR
1991	9	30	-100.73	31.85			2.20	M	230.5	143.2	NMTR
1991	10	5	-105.41	31.38			2.20	M	248.6	154.5	NMTR
1992	1	2	-103.19	32.30			5.00	M	17.8	11.0	NMTR
1992	1	2	-103.19	32.30			1.80	M	17.8	11.0	NMTR
1992	1	2	-103.19	32.30			1.50	M	17.8	11.0	NMTR
1992	1	2	-103.19	32.30			2.40	M	17.8	11.0	NMTR
1992	1	2	-103.19	32.30			1.80	M	17.8	11.0	NMTR
1992	1	3	-103.19	32.30			1.90	M	17.8	11.0	NMTR
1992	1	4	-103.19	32.30			1.50	M	17.8	11.0	NMTR
1992	1	7	-103.19	32.30			2.40	M	17.8	11.0	NMTR
1992	1	9	-103.19	32.30			2.80	M	17.8	11.0	NMTR

Table 3.2-20 Location of Recorded Earthquakes Within a 322 km (200 mi) Radius of the NEF Site

Year	Month	Day	Longitude	Latitude	Focal Depth[1]		MAG[2]	MAG Type[3]	Epicentral Distance		Data Sources[4]
			(°W)	(°N)	(km)	(mi)			(km)	(mi)	
1992	1	11	-103.19	32.30			2.00	M	17.8	11.0	NMTR
1992	1	23	-102.29	31.84			1.90	M	99.2	61.7	NMTR
1992	2	2	-102.86	32.17			1.90	M	36.4	22.6	NMTR
1992	3	15	-104.12	34.92			1.70	M	292.1	181.5	NMTR
1992	3	28	-105.39	33.45			1.80	M	242.2	150.5	NMTR
1992	4	3	-103.03	32.26			2.10	M	19.9	12.4	NMTR
1992	4	6	-102.61	31.86			1.70	M	77.7	48.3	NMTR
1992	4	7	-102.29	31.56			1.60	M	122.6	76.2	NMTR
1992	4	7	-102.29	31.56			2.30	M	122.6	76.2	NMTR
1992	4	7	-102.29	31.56			1.70	M	122.6	76.2	NMTR
1992	4	8	-104.86	32.41			1.60	M	166.9	103.7	NMTR
1992	4	30	-104.31	30.66			1.70	M	229.0	142.3	NMTR
1992	5	9	-104.34	30.49			1.60	M	246.7	153.3	NMTR
1992	5	15	-103.08	32.28			1.60	M	17.5	10.9	NMTR
1992	5	16	-102.34	31.75			1.70	M	103.0	64.0	NMTR
1992	6	14	-103.10	32.30			2.30	M	15.1	9.4	NMTR
1992	6	20	-102.42	31.43			1.60	M	127.5	79.2	NMTR
1992	6	20	-102.42	31.43			1.50	M	127.5	79.2	NMTR
1992	6	29	-102.47	31.42			1.40	M	126.9	78.8	NMTR
1992	6	29	-102.47	31.42			1.40	M	126.9	78.8	NMTR
1992	6	29	-102.47	31.42			2.00	M	126.9	78.8	NMTR
1992	7	5	-102.39	31.88			1.50	M	89.4	55.6	NMTR
1992	7	5	-102.39	31.88			1.30	M	89.4	55.6	NMTR
1992	7	21	-103.13	32.28			1.90	M	17.8	11.1	NMTR
1992	8	12	-102.41	31.39			1.50	M	131.9	82.0	NMTR
1992	8	18	-102.45	31.46			1.90	M	123.5	76.7	NMTR
1992	8	19	-100.92	33.11			2.20	M	215.3	133.8	NMTR
1992	8	26	-102.71	32.17	5.0	3.1	3.00	un	45.6	28.4	ANSS
1992	8	28	-100.98	32.38			1.70	M	197.4	122.6	NMTR
1992	9	4	-102.26	31.42			1.90	M	136.8	85.0	NMTR
1992	9	15	-103.02	32.16			2.20	M	31.6	19.6	NMTR
1992	10	8	-102.81	32.25			1.60	M	33.1	20.6	NMTR
1992	10	10	-102.41	31.71			1.60	M	102.2	63.5	NMTR
1992	10	27	-101.93	34.12			1.30	M	215.1	133.7	NMTR
1992	11	22	-103.16	32.29			1.70	M	18.0	11.2	NMTR
1992	11	27	-102.49	31.44			1.30	M	124.0	77.1	NMTR
1992	12	2	-102.35	31.42			2.40	M	131.5	81.7	NMTR
1992	12	3	-103.74	33.66			1.90	M	149.6	93.0	NMTR
1992	12	5	-102.51	31.87			1.40	M	83.0	51.6	NMTR
1993	1	4	-105.27	31.06			1.30	M	256.5	159.4	NMTR
1993	1	28	-102.58	31.85			1.80	M	80.3	49.9	NMTR
1993	1	31	-104.64	30.60			1.50	M	250.8	155.9	NMTR
1993	2	11	-105.23	31.12			2.00	M	250.1	155.4	NMTR
1993	2	28	-102.43	31.21			1.30	M	149.4	92.8	NMTR
1993	2	28	-102.41	31.22			1.50	M	149.3	92.8	NMTR
1993	3	8	-103.33	30.87			1.60	M	175.9	109.3	NMTR

Year	Month	Day	Longitude	Latitude	Focal Depth[1]		MAG[2]	MAG Type[3]	Epicentral Distance		Data Sources[4]
			(°W)	(°N)	(km)	(mi)			(km)	(mi)	
1993	3	21	-102.37	31.43			1.50	M	130.4	81.0	NMTR
1993	4	23	-102.47	31.21			1.70	M	147.8	91.9	NMTR
1993	5	5	-105.16	32.29			2.10	M	195.3	121.4	NMTR
1993	5	16	-105.06	30.44			2.20	M	290.1	180.2	NMTR
1993	5	17	-102.33	31.42			2.30	M	133.3	82.9	NMTR
1993	5	23	-102.42	31.42			1.60	M	128.7	80.0	NMTR
1993	5	28	-103.12	32.75			2.50	M	34.6	21.5	NMTR
1993	6	17	-102.56	31.80			1.70	M	86.5	53.8	NMTR
1993	6	23	-102.44	31.51			1.40	M	119.5	74.2	NMTR
1993	6	23	-102.54	31.43			2.50	M	123.2	76.6	NMTR
1993	6	23	-102.52	31.43			2.80	M	123.2	76.5	NMTR
1993	6	23	-102.52	31.43			2.10	M	123.2	76.5	NMTR
1993	6	23	-102.54	29.66			1.90	M	312.3	194.0	NMTR
1993	6	23	-102.51	31.35	5.0	3.1	2.80	un	132.5	82.3	ANSS
1993	6	24	-102.45	31.48			2.10	M	121.9	75.7	NMTR
1993	7	3	-102.43	31.44			1.50	M	126.7	78.7	NMTR
1993	7	3	-102.34	31.50			2.20	M	125.5	78.0	NMTR
1993	7	3	-102.38	31.54			1.60	M	119.3	74.1	NMTR
1993	8	13	-102.52	31.89			1.30	M	80.1	49.8	NMTR
1993	8	29	-102.91	32.35			2.50	M	19.0	11.8	NMTR
1993	9	5	-100.96	32.28			2.00	M	200.1	124.4	NMTR
1993	9	6	-100.91	32.48			1.80	M	203.6	126.5	NMTR
1993	9	11	-103.76	34.72			1.50	M	260.9	162.1	NMTR
1993	9	26	-103.52	35.08			1.50	M	296.6	184.3	NMTR
1993	9	30	-103.80	33.64			1.90	M	149.0	92.6	NMTR
1993	10	3	-103.84	33.61			1.70	M	148.5	92.3	NMTR
1993	11	6	-102.19	31.75			1.50	M	113.6	70.6	NMTR
1993	11	24	-104.74	32.34			1.30	M	156.2	97.1	NMTR
1993	11	25	-102.10	34.27			2.60	M	223.0	138.5	NMTR
1993	11	25	-104.38	30.49			1.30	M	248.6	154.5	NMTR
1993	12	2	-102.34	31.27			1.30	M	147.3	91.5	NMTR
1993	12	3	-102.23	31.68			1.60	M	115.6	71.8	NMTR
1993	12	10	-102.29	31.74			1.60	M	106.8	66.4	NMTR
1993	12	18	-103.41	30.21			1.80	M	249.5	155.0	NMTR
1993	12	22	-105.68	33.33	10.0	6.2	3.20	un	261.9	162.8	ANSS
1994	1	6	-105.09	31.95			2.40	M	196.3	122.0	NMTR
1994	1	7	-102.32	31.24			1.70	M	151.0	93.8	NMTR
1994	3	15	-103.56	30.11			2.00	M	261.9	162.8	NMTR
1994	4	21	-103.12	32.31			1.40	M	14.1	8.8	NMTR
1994	4	25	-104.62	30.60			1.90	M	250.5	155.7	NMTR
1994	5	23	-102.64	32.11			1.60	M	55.0	34.2	NMTR
1994	6	30	-102.33	31.36			1.30	M	138.6	86.2	NMTR
1994	8	22	-102.21	33.34			1.60	M	129.0	80.2	NMTR
1994	8	30	-102.32	31.38			1.40	M	137.3	85.3	NMTR
1994	8	30	-102.32	31.34			1.50	M	141.5	87.9	NMTR
1994	8	30	-102.30	31.42			1.30	M	135.1	84.0	NMTR

Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth[1] (km)	Focal Depth[1] (mi)	MAG[2]	MAG Type[3]	Epicentral Distance (km)	Epicentral Distance (mi)	Data Sources[4]
1994	9	24	-102.36	31.43			2.00	M	131.1	81.4	NMTR
1994	11	24	-100.80	32.39			2.70	M	214.3	133.2	NMTR
1995	1	1	-102.45	31.77			1.40	M	94.7	58.8	NMTR
1995	1	4	-102.38	31.48			1.30	M	125.0	77.6	NMTR
1995	2	1	-104.09	34.51			1.80	M	248.7	154.6	NMTR
1995	3	19	-104.21	35.00	5.0	3.1	3.30	un	303.1	188.4	ANSS
1995	4	14	-103.35	30.28			5.70	M	240.7	149.5	UTIG
1995	4	18	-102.27	31.44			1.90	M	134.5	83.6	NMTR
1995	4	18	-105.34	31.10			1.60	M	259.8	161.4	NMTR
1995	4	21	-103.35	30.30	10.0	6.2	2.90	un	238.5	148.2	ANSS
1995	5	11	-105.20	32.71			2.40	M	200.4	124.5	NMTR
1995	5	15	-102.42	31.40			1.80	M	131.1	81.5	NMTR
1995	5	27	-102.34	31.34			2.30	M	140.1	87.0	NMTR
1995	5	30	-105.21	32.71			2.10	M	200.9	124.8	NMTR
1995	7	11	-105.06	30.87			1.80	M	255.5	158.8	NMTR
1995	7	17	-104.94	31.15			1.40	M	226.0	140.4	NMTR
1995	8	1	-105.27	33.14			1.30	M	218.9	136.0	NMTR
1995	8	2	-103.36	30.31			1.80	M	237.2	147.4	NMTR
1995	8	12	-103.07	30.79			1.90	M	183.1	113.8	NMTR
1995	8	14	-102.96	30.41			1.50	M	225.3	140.0	NMTR
1995	10	19	-104.84	32.05			2.00	M	170.4	105.9	NMTR
1995	10	25	-103.42	30.35			2.20	M	233.6	145.2	NMTR
1995	11	12	-103.35	30.30	10.0	6.2	3.60	ML	238.5	148.2	ANSS
1995	12	3	-104.90	31.93			1.50	M	180.1	111.9	NMTR
1995	12	4	-104.90	31.93			1.40	M	180.1	111.9	NMTR
1995	12	4	-104.90	31.93			1.30	M	180.1	111.9	NMTR
1996	3	15	-105.69	33.59	10.0	6.2	2.90	ML	274.6	170.6	ANSS
1998	4	15	-103.30	30.19	10.0	6.2	3.60	ML	250.4	155.6	ANSS
1999	3	1	-104.66	32.57	1.0	0.6	2.90	ML	148.1	92.0	ANSS
1999	3	14	-104.63	32.59	1.0	0.6	4.00	ML	145.9	90.7	ANSS
1999	3	17	-104.67	32.58	1.0	0.6	3.50	Mc	149.7	93.0	ANSS
1999	5	30	-104.66	32.58	10.0	6.2	3.90	ML	148.9	92.5	ANSS
1999	8	9	-104.59	32.57	5.0	3.1	2.90	Mc	142.0	88.3	ANSS
2000	2	2	-104.63	32.58	5.0	3.1	2.70	ML	145.7	90.5	ANSS
2000	2	26	-103.61	30.24	5.0	3.1	2.80	ML	248.6	154.5	ANSS
2001	6	2	-103.14	32.33	5.0	3.1	3.30	ML	12.6	7.8	ANSS
2001	11	22	-102.63	31.79	5.0	3.1	3.10	ML	83.7	52.0	ANSS
2002	9	17	-104.63	32.58	10.0	6.2	3.50	ML	145.8	90.6	ANSS
2002	9	17	-104.63	32.58	10.0	6.2	3.30	ML	145.8	90.6	ANSS
2003	6	21	-104.51	32.67	5.0	3.1	3.60	ML	135.5	84.2	ANSS

Notes:

[1] Focal depth information only available for events reported in ANSS Catalog
[2] MAG - Magnitude
[3] MAG Type
- M – Moment Magnitude
- mb – Body – wave Magnitude
- un – Unspecified Magnitude
- ML – Local Magnitude
- Mc – Coda – wave Magnitude

[4] Data Sources
- UTIG – University of Texas Institute for Geophysics
- NMTH – New Mexico Tech Historical Catalog
- NMTR – New Mexico Tech Regional Catalog, Exclusive of Socorro NM Events
- ANSS – Advanced National Seismic System

Table 3.2-21 Earthquakes of Magnitude 3.0 and Greater Within 322 km (200 mi) Radius of the NEF Site

NEF Site Coordinates			Longitude	Latitude							
			103.0820	32.4360							
Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal (km)	Depth[1] (mi)	MAG[2]	MAG Type[3]	Epicentral Distance (km)	Epicentral Distance (mi)	Data Sources[4]
1931	8	16	-104.60	30.70			6.00	M	240.3	149.3	UTIG
1949	5	23	-105.20	34.60			4.50	M	310.0	192.6	NMTH
1955	1	27	-104.50	30.60			3.30	M	244.0	151.6	UTIG
1962	3	6	-104.80	31.20			3.50	M	212.3	131.9	UTIG
1963	12	19	-104.27	34.82			3.40	M	287.0	178.3	NMTR
1964	11	8	-103.10	31.90			3.00	M	59.5	37.0	UTIG
1964	11	21	-103.10	31.90			3.10	M	59.5	37.0	UTIG
1965	2	3	-103.10	31.90			3.30	M	59.5	37.0	UTIG
1965	8	30	-103.00	31.90			3.50	M	60.0	37.3	UTIG
1966	8	14	-103.00	31.90			3.40	M	60.0	37.3	UTIG
1966	11	26	-105.44	30.95			3.50	M	277.5	172.4	NMTR
1971	7	30	-103.00	31.72	10.0	6.2	3.00	mb	79.9	49.6	ANSS
1971	7	31	-103.06	31.70	10.0	6.2	3.40	mb	81.4	50.6	ANSS
1971	9	24	-103.20	31.60			3.20	M	93.5	58.1	UTIG
1972	7	26	-104.01	32.57			3.10	M	88.3	54.9	NMTR
1973	8	2	-105.56	31.04			3.60	M	280.7	174.5	NMTR
1973	8	4	-103.22	35.11			3.00	M	296.6	184.3	NMTR
1974	11	28	-104.14	32.31	5.0	3.1	3.90	mb	100.4	62.4	ANSS
1974	12	30	-103.10	30.90			3.70	M	170.5	106.0	UTIG
1975	2	2	-103.19	35.05			3.00	M	290.7	180.6	NMTR
1975	8	1	-104.00	31.40			3.00	M	143.9	89.4	UTIG
1975	12	12	-102.31	31.61			3.00	M	117.5	73.0	NMTR
1976	1	19	-103.09	31.90			3.50	M	59.5	37.0	UTIG
1976	1	25	-103.08	31.90	2.0	1.2	3.90	un	59.3	36.8	ANSS
1976	8	5	-103.00	31.60			3.00	M	93.1	57.9	UTIG
1976	9	17	-102.50	31.40			3.10	M	127.4	79.2	UTIG
1977	4	26	-103.08	31.90	4.0	2.5	3.30	un	59.3	36.8	ANSS
1977	6	7	-100.75	33.06	5.0	3.1	4.00	un	228.5	142.0	ANSS
1977	7	22	-102.70	31.80			3.00	M	79.2	49.2	UTIG
1977	11	28	-100.84	32.95	5.0	3.1	3.50	un	217.4	135.1	ANSS
1978	3	2	-102.38	31.58			3.30	M	115.4	71.7	NMTR
1978	3	2	-102.56	31.55			3.50	M	109.9	68.3	UTIG
1978	6	16	-100.80	33.00			3.40	M	222.1	138.0	UTIG
1978	6	16	-100.77	33.03	10.0	6.2	5.30	un	226.1	140.5	ANSS
1978	6	29	-102.42	31.08			3.20	M	163.1	101.4	NMTR
1982	1	4	-102.49	31.18	5.0	3.1	3.90	un	149.9	93.2	ANSS
1982	11	28	-100.84	33.00	5.0	3.1	3.30	un	218.4	135.7	ANSS
1983	9	15	-104.43	34.92			3.10	M	302.6	188.1	NMTR
1984	5	21	-102.23	35.07	5.0	3.1	3.10	un	302.5	188.0	ANSS
1984	9	11	-100.70	31.99	5.0	3.1	3.20	un	229.4	142.5	ANSS

Table 3.2-21 Earthquakes of Magnitude 3.0 and Greater Within 322 km (200 mi) Radius of the NEF Site

Year	Month	Day	Longitude	Latitude	Focal	Depth[1]	MAG[2]	MAG Type[3]	Epicentral Distance		Data Sources[4]
			(°W)	(°N)	(km)	(mi)			(km)	(mi)	
1984	9	19	-100.69	32.03	5.0	3.1	3.00	un	229.3	142.5	ANSS
1986	1	30	-100.69	32.07	5.0	3.1	3.30	un	228.0	141.7	ANSS
1990	8	3	-100.69	32.21			3.40	M	225.6	140.2	NMTR
1992	1	2	-103.19	32.30			5.00	M	17.8	11.0	NMTR
1992	8	26	-102.71	32.17	5.0	3.1	3.00	un	45.6	28.4	ANSS
1993	12	22	-105.68	33.33	10.0	6.2	3.20	un	261.9	162.8	ANSS
1995	3	19	-104.21	35.00	5.0	3.1	3.30	un	303.1	188.4	ANSS
1995	4	14	-103.35	30.28			5.70	M	240.7	149.5	UTIG
1995	11	12	-103.35	30.30	10.0	6.2	3.60	ML	238.5	148.2	ANSS
1998	4	15	-103.30	30.19	10.0	6.2	3.60	ML	250.4	155.6	ANSS
1999	3	14	-104.63	32.59	1.0	0.6	4.00	ML	145.9	90.7	ANSS
1999	3	17	-104.67	32.58	1.0	0.6	3.50	Mc	149.7	93.0	ANSS
1999	5	30	-104.66	32.58	10.0	6.2	3.90	ML	148.9	92.5	ANSS
2001	6	2	-103.14	32.33	5.0	3.1	3.30	ML	12.6	7.8	ANSS
2001	11	22	-102.63	31.79	5.0	3.1	3.10	ML	83.7	52.0	ANSS
2002	9	17	-104.63	32.58	10.0	6.2	3.50	ML	145.8	90.6	ANSS
2002	9	17	-104.63	32.58	10.0	6.2	3.30	ML	145.8	90.6	ANSS
2003	6	21	-104.51	32.67	5.0	3.1	3.60	ML	135.5	84.2	ANSS

Notes:

[1] Focal depth information only available for events reported in ANSS Catalog
[2] MAG - Magnitude
[3] MAG Type
- M – Moment Magnitude
- mb – Body – wave Magnitude
- un – Unspecified Magnitude
- ML – Local Magnitude
- Mc – Coda – wave Magnitude

[4] Data Sources
- UTIG – University of Texas Institute for Geophysics
- NMTH – New Mexico Tech Historical Catalog
- NMTR – New Mexico Tech Regional Catalog, Exclusive of Socorro NM Events
- ANSS – Advanced National Seismic System

Table 3.2-22 Earthquake Data Sources for New Mexico and West Texas
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Data Source	**Time Span**	**Number of events in 322 km (200 mi) Radius**
New Mexico Tech, Regional Catalog	1962 - 1995	504
New Mexico Tech, Historical Catalog	1869 - 1992	2
University of Texas Institute of Geophysics	1931 - 1998	42
Advanced National Seismic System	1962 - 2003	64

Table 3.2-23 Modified Mercalli Intensity Scale
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Intensity Value	Description
I	Not felt except by a very few under especially favorable circumstances.
II	Felt only by a few persons at rest, especially on upper floors of buildings. Delicately suspended objects may swing.
III	Felt quite noticeably indoors, especially on upper floors of buildings, but many people do not recognize it as an earthquake. Standing automobiles may rock slightly. Vibration like passing of truck.
IV	During the day felt indoors by many, outdoors by few. At night some awakened. Dishes, windows, doors disturbed; walls make creaking sound. Sensation like heavy truck striking building. Standing automobiles rocked noticeably.
V	Felt by nearly everyone, many awakened. Some dishes, windows, and so on broken; cracked plaster in a few places; unstable objects overturned. Disturbances of trees, poles, and other tall objects sometimes noticed. Pendulum clocks may stop.
VI	Felt by all, many frightened and run outdoors. Some heavy furniture moved; a few instances of fallen plaster and damaged chimneys. Damage slight.
VII	Everybody runs outdoors. Damage negligible in buildings of good design and construction; slight to moderate in well-built ordinary structures; considerable in poorly built or badly designed structures; some chimneys broken. Noticed by persons driving cars.
VIII	Damage slight in specially designed structures; considerable in ordinary substantial buildings, with partial collapse; great in poorly built structures. Panel walls thrown out of frame structures. Fall of chimneys, factory stacks, columns, monuments, walls. Heavy furniture overturned. Sand and mud ejected in small amounts. Changes in well water. Persons driving cars disturbed.
IX	Damage considerable in specially designed structures; well-designed frame structures thrown out of plumb; great in substantial buildings, with partial collapse. Buildings shifted off foundations. Ground cracked conspicuously. Underground pipes broken.
X	Some well-built wooden structures destroyed; most masonry and frame structures destroyed with foundations; ground badly cracked. Rails bent. Landslides considerable from river banks and steep slopes. Shifted sand and mud. Water splashed, slopped over banks.
XI	Few, if any (masonry) structures remain standing. Bridges destroyed. Broad fissures in ground. Underground pipelines completely out of service. Earth slumps and land slips in soft ground. Rails bent greatly.
XII	Damage total. Waves seen on ground surface. Lines of sight and level distorted. Objects thrown in the air.

Table 3.2-24 Comparison of Parameters for the January 2, 1992 Eunice, New Mexico Earthquake

Year	Month	Day	Longitude	Latitude	Magnitude	Data Source[1]
1992	1	2	-103.1863	32.3025	5.0	NMTR
1992	1	2	-102.97	32.36	4.6	UTIG
1992	1	2	-103.2	32.3	5.0	NMTH
1992	1	2	-103.101	32.336	5.0	ANSS

[1]Data Sources:

UTIG	University of Texas Institute for Geophysics
NMTH	New Mexico Tech Historical Catalog
ANSS	Advanced National Seismic System
NMTR	New Mexico Tech Regional Catalog, exclusive of Socorro New Mexico events

Table 3.2-25 Earthquake Recurrence Models for the NEF Site Region

Earthquake Recurrence Models

Zone	Area (km^2)		a-value	b-value	Beta	Rate/yr M > = 5.0	Return Period M > = 5.0
200 Mile Radius	253,502	best fit	2.15	-0.74	-1.704	0.0282	35
		fixed b, -0.9	2.80	-0.90	-2.072	0.0200	50
Region 1 – 100 Mile Radius	78,758	best fit	2.25	-0.89	-2.049	0.0063	158
		fixed b, -0.9	2.40	-0.90	-2.072	0.0079	126
Central Basin Earthquake Cluster	15,065	best fit	1.98	-0.86	-1.980	0.0048	209
		fixed b, -0.9	2.20	-0.90	-2.072	0.0050	200

Table 3.2-26 Earthquake Recurrence Models for the Central Basin Platform (CBP)in the Waste Isolation Pilot Project (WIPP) Safety Analysis Report (SAR)

WIPP SAR Earthquake Recurrence Models

Zone	Area (km^2)	a-value	b-value	Beta	Rate/yr M > = 5.0	Return Period M > = 5.0
WIPP SAR						
Background	10,000 M uncorrected	1.439	-1.000	2.303	0.0003	3639
Background	10,000 M corrected	1.939	-1.000	2.303	0.0009	1151
Rio Grande Rift	110,000 M uncorrected	2.560	-1.000	2.303	0.0036	275
Rio Grande Rift	110,000 M corrected	3.060	-1.000	2.303	0.0115	87
Basin & Range Subregion	640,000 M uncorrected	2.750	-1.000	2.303	0.0056	178
Basin & Range Subregion	640,000 M corrected	3.250	-1.000	2.303	0.0178	56
WIPP Central Basin Platform	7,500 M uncorrected	2.740	-0.900	2.072	0.0174	58
WIPP Central Basin Platform	7,500 M corrected	3.190	-0.900	2.072	0.0490	20

Table 3.2-27 Attenuation Model Formulas and Coefficients

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Model	Ground Motion Parameter (y)	c_1	c_2	c_3	c_4			
EPRI, 1988	psrv (1 Hz)	-7.95	2.14	-1.00	-0.0018			
Hard Rock Site Condition	psrv (2.5 Hz)	-3.82	1.49	-1.00	-0.0024			
$\sigma_{\ln(y)} = 0.5$	psrv (5 Hz)	-2.11	1.20	-1.00	-0.0031			
	psrv (10 Hz)	-1.55	1.05	-1.00	-0.0039			
	psrv (25 Hz)	-1.63	0.98	-1.00	-0.0053			
	PGA	2.55	1.00	-1.00	-0.0046			
Equation:	$\ln(y) = c_1 + c_2 m_{Lg} + c_3 \ln(R) + c_4 R$							
Nuttli, 1986	psrv (1 Hz)†	0.29	1.15	-0.83	-0.0028			
Firm Rock Site Condition	psrv (2.5 Hz)†	-0.62	1.15	-0.83	-0.0028			
$\sigma_{\ln(y)} = 0.5$	psrv (5 Hz)†	-1.32	1.15	-0.83	-0.0028			
	psrv (10 Hz)†	-2.13	1.15	-0.83	-0.0028			
	psrv (25 Hz)†	-3.53	1.15	-0.83	-0.0028			
	PGA	1.38	1.15	-0.83	-0.0028			
Equations:	**† For a given m_{Lg} and R, ln(y) is the smaller of:** $c_1 + c_2 m_{Lg} + c_3 \ln R + c_4 R$ **and,** $-8.3 + 2.3 m_{Lg} - 0.83 \ln(R) - 0.0012R$							
		c_1	c_2	c_3	c_4	c_5	c_6	c_7
Toro, 1997	Sa (0.5 Hz)	-0.74	1.86	-0.31	0.92	0.46	0.0017	6.9
Midcontinent,	Sa (1 Hz)	0.09	1.42	-0.20	0.90	0.49	0.0023	6.8
Moment magnitude scaling	Sa (2.5 Hz)	1.07	1.05	-0.10	0.93	0.56	0.0033	7.1
	Sa (5 Hz)	1.73	0.84	0	0.98	0.66	0.0042	7.5
	Sa (10 Hz)	2.37	0.81	0	1.10	1.02	0.0040	8.3
	Sa (25 Hz)	3.68	0.80	0	1.46	1.77	0.0013	10.5
	Sa (35 Hz)	4.00	0.79	0	1.57	1.83	0.0008	11.1
	PGA	2.20	0.81	0	1.27	1.16	0.0021	9.3
Equations:	$\ln(y) = c_1 + c_2(M-6) + c_3(M-6)^2 - c_4 \ln(R_M) - (c_5-c_4)\max[\ln(R_M/100),0] - c_6 R_M + \varepsilon_U + \varepsilon_r$; $R_M = (R^2 + c_7^2)^{1/2}$							

Note: psrv = pseudo relative velocity at given frequency
PGA = peak ground acceleration
Sa = Spectral acceleration at given frequency

Table 3.2-28 Seismic Hazard Results at NEF Site From Rio Grande Rift Seismic Source Zones

Page 1 of 1

cm/s^2	(g)	WIPP Basin and Range	WIPP Rio Grande Rift	WIPP M corr Basin and Range	WIPP M corr Rio Grande Rift
peak ground accel.		**Annual probability of PGA being exceeded**			
4.94	0.005	4.45E-03	2.78E-03		
9.81	0.010	2.29E-03	1.35E-03	7.26E-03	4.31E-03
49.01	0.050	4.84E-05	2.42E-05	1.54E-04	7.74E-05
73.55	0.075	1.08E-05	5.09E-06	3.44E-05	1.63E-05
98.10	0.100	3.13E-06	1.39E-06	9.95E-06	4.46E-06
122.61	0.125	1.06E-06	4.52E-07	3.38E-06	1.45E-06
147.08	0.150	4.05E-07	1.65E-07	1.29E-06	5.28E-07
196.17	0.200	7.41E-08	2.81E-08	2.36E-07	8.98E-08
245.18	0.250	1.70E-08	6.08E-09	5.40E-08	1.94E-08
294.12	0.300	4.59E-09	1.56E-09	1.46E-08	4.98E-09
392.29	0.400	4.68E-10	1.46E-10	1.49E-09	4.67E-10
490.29	0.500	6.61E-11	1.92E-11	2.10E-10	6.14E-11

Table 3.2-29 Seismic Hazard Results at NEF Site From Local Source Zones

PGA (g)	B100B9W Mx=6.0	B100BFW Mx=6.0	B200B9W Mx=6.5	B200BFW Mx=6.5	Bk53B9W Mx=5.25	Bk53BFW Mx=5.25	B260B9W Mx=6.0	B260BFW Mx=6.0	Bk53B9T Mx=5.25	Bk53BFT Mx=5.25	B260B9T Mx-=6.0	B260BFT Mx=6.0	Weighted Average
	Annual Probability of PGA Being Exceeded												
0.010	8.09E-03	7.21E-03	1.32E-02	1.91E-02	7.66E-03	6.83E-03	1.26E-02	1.81E-02	4.97E-03	4.45E-03	4.72E-03	6.87E-03	8.88E-03
0.050	1.69E-03	1.54E-03	1.27E-03	1.99E-03	1.09E-03	9.93E-04	9.74E-04	1.45E-03	5.65E-04	5.15E-04	4.18E-04	6.17E-04	1.01E-03
0.075	8.30E-04	7.60E-04	5.61E-04	8.88E-04	4.99E-04	4.55E-04	4.20E-04	6.26E-04	2.67E-04	2.43E-04	2.00E-04	2.97E-04	4.62E-04
0.100	4.75E-04	4.36E-04	3.07E-04	4.87E-04	2.69E-04	2.46E-04	2.26E-04	3.38E-04	1.43E-04	1.31E-04	1.13E-04	1.68E-04	2.53E-04
0.125	2.97E-04	2.74E-04	1.88E-04	3.01E-04	1.58E-04	1.45E-04	1.37E-04	2.05E-04	8.21E-05	7.50E-05	6.97E-05	1.04E-04	1.52E-04
0.150	1.97E-04	1.82E-04	1.25E-04	2.00E-04	9.81E-05	8.97E-05	8.89E-05	1.34E-04	4.91E-05	4.49E-05	4.55E-05	6.85E-05	9.76E-05
0.200	9.59E-05	8.88E-05	6.25E-05	1.02E-04	4.12E-05	3.77E-05	4.25E-05	6.45E-05	1.90E-05	1.73E-05	2.15E-05	3.26E-05	4.44E-05
0.250	5.12E-05	4.75E-05	3.51E-05	5.77E-05	1.87E-05	1.71E-05	2.26E-05	3.45E-05	7.89E-06	7.21E-06	1.11E-05	1.70E-05	2.21E-05
0.300	2.91E-05	2.70E-05	2.12E-05	3.53E-05	8.93E-06	8.17E-06	1.28E-05	1.98E-05	3.44E-06	3.15E-06	6.04E-06	9.38E-06	1.17E-05
0.400	1.06E-05	9.84E-06	8.85E-06	1.51E-05	2.23E-06	2.04E-06	4.66E-06	7.29E-06	7.00E-07	6.39E-07	2.02E-06	3.20E-06	3.64E-06
0.500	4.32E-06	4.03E-06	4.20E-06	7.32E-06	5.87E-07	5.35E-07	1.89E-06	3.00E-06	1.40E-07	1.27E-07	7.53E-07	1.21E-06	1.23E-06

Notes:

PGA = Peak horizontal ground acceleration in firm rock
W = WIPP attenuation model; T = Toro et al. (1997) approx. model
Mx = Maximum magnitude

Table 3.2-30 Peak Acceleration Seismic Hazard Summary for the NEF Site

Seismic Source	250 – year earthquake PGA as % g	475 – year earthquake PGA as % g
Local seismic zones	2.4%	3.6%
Max. for Rio Grande Rift	1.0%	1.8%

Table 3.2-31 Regulatory Guide 1.60 Ratio of Vertical to Horizontal Component Design Response Spectra

Page 1 of 1

Period range	Ratio Vertical/Horizontal
> 4.0 s (< 0.25 Hz)	2/3
< 0.29 s (> 3.5 Hz)	1.0
Between 0.29 and 4.0 s	Varies between 2/3 and 1.0

Table 3.2-32 Horizontal Response Spectrum for the 10,000-Year Design Earthquake
Page 1 of 1

Soil Class C			
Period s	**psrv cm/s**	**Sa (g)**	**SD mm**
0.020	0.472	0.151	0.015
0.030	0.715	0.151	0.034
0.040	1.420	0.227	0.090
0.100	5.473	0.351	0.871
0.200	10.809	0.346	3.440
0.400	10.809	0.173	6.881
1.000	10.809	0.069	17.202
2.000	5.404	0.017	17.202

psrv = pseudo relative velocity
Sa = spectral acceleration
SD = spectral displacement

Table 3.2-33 Vertical Response Spectrum for the 10,000-Year Design Earthquake
Page 1 of 1

Soil Class C			
Period s	**psrv cm/s**	**Sa (g)**	**SD mm**
0.020	0.472	0.151	0.015
0.030	0.715	0.151	0.034
0.040	1.420	0.227	0.090
0.100	5.473	0.351	0.871
0.200	7.242	0.232	2.305
0.400	7.242	0.116	4.610
1.000	7.242	0.046	11.526
2.000	3.621	0.012	11.526

psrv = pseudo relative velocity
Sa = spectral acceleration
SD = spectral displacement

FIGURES

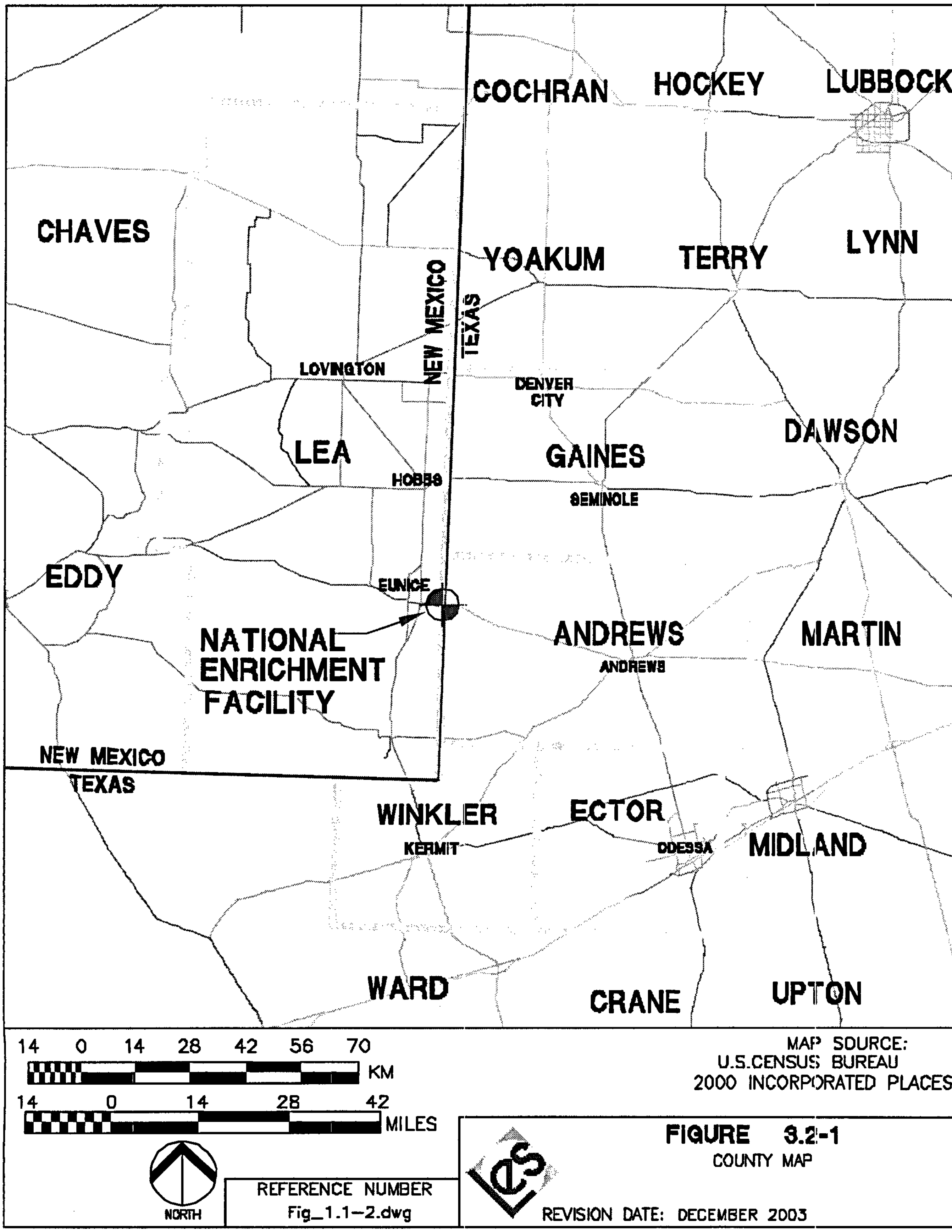
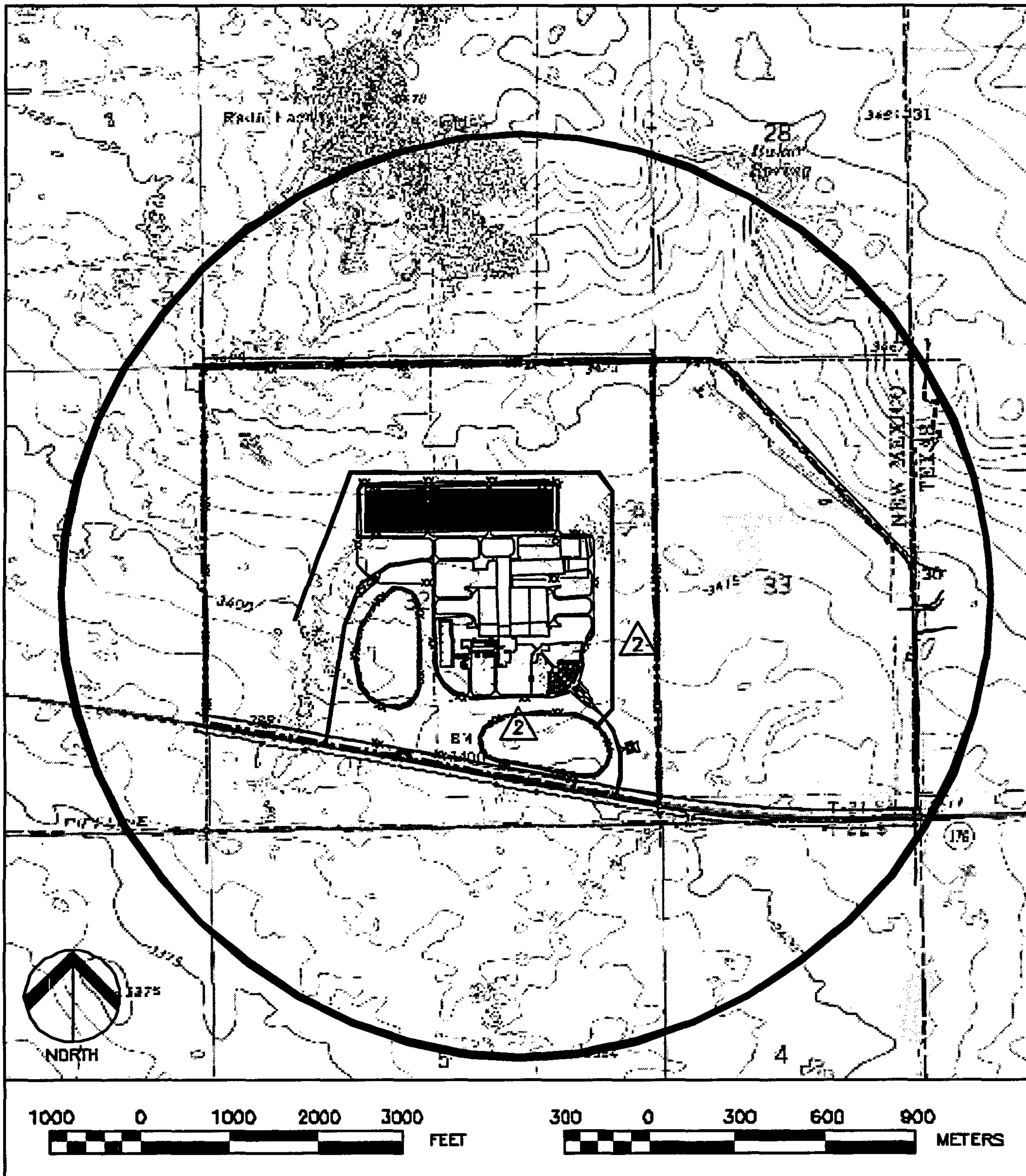


FIGURE 3.2-1
COUNTY MAP

REVISION DATE: DECEMBER 2003



MAP SOURCE:
USGS 7.5 MINUTE
EUNICE NE QUADRANGLE
TEX.–N. MEX. 1:24000

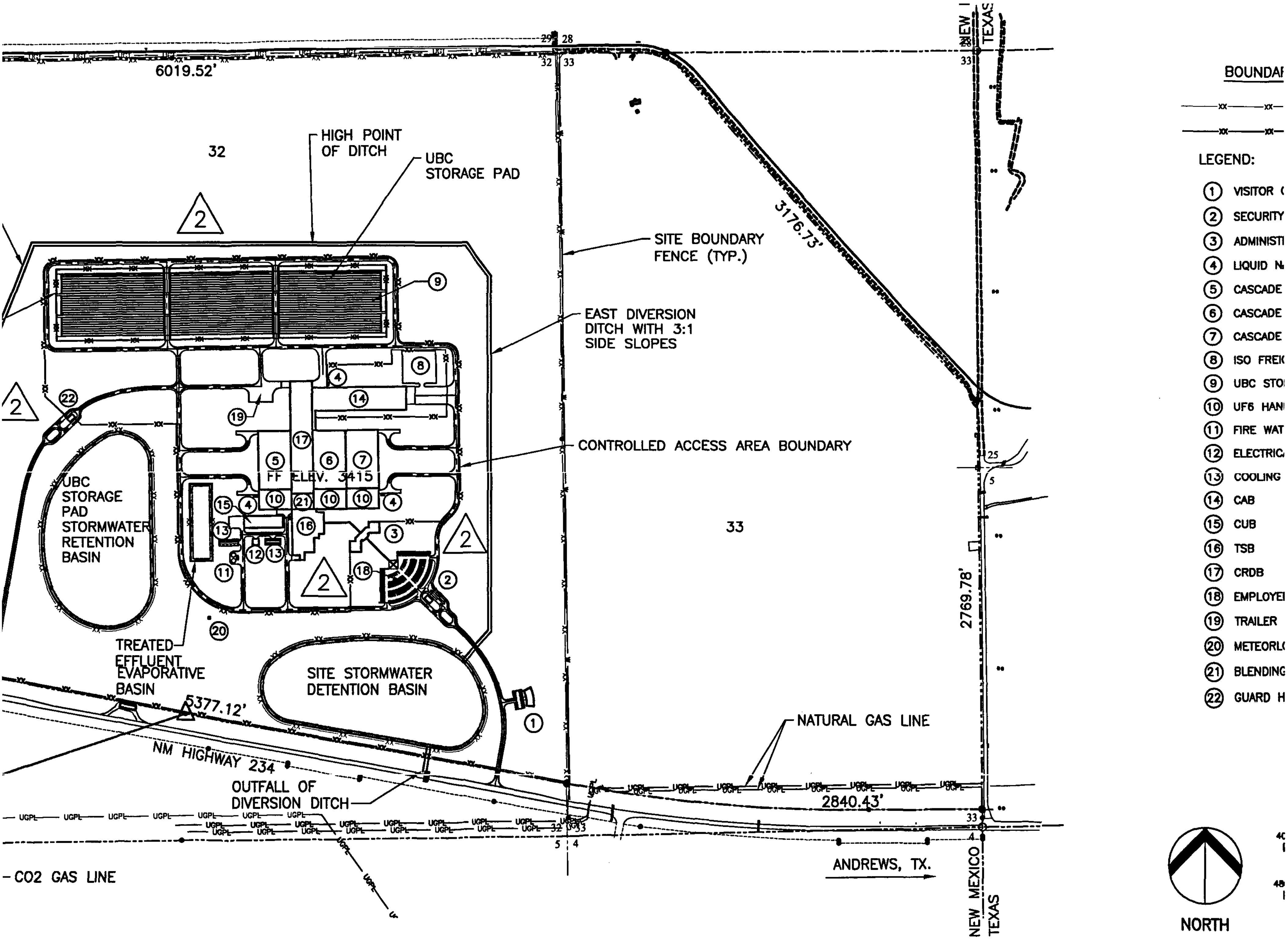
CONTOUR INTERVAL:
5 FEET

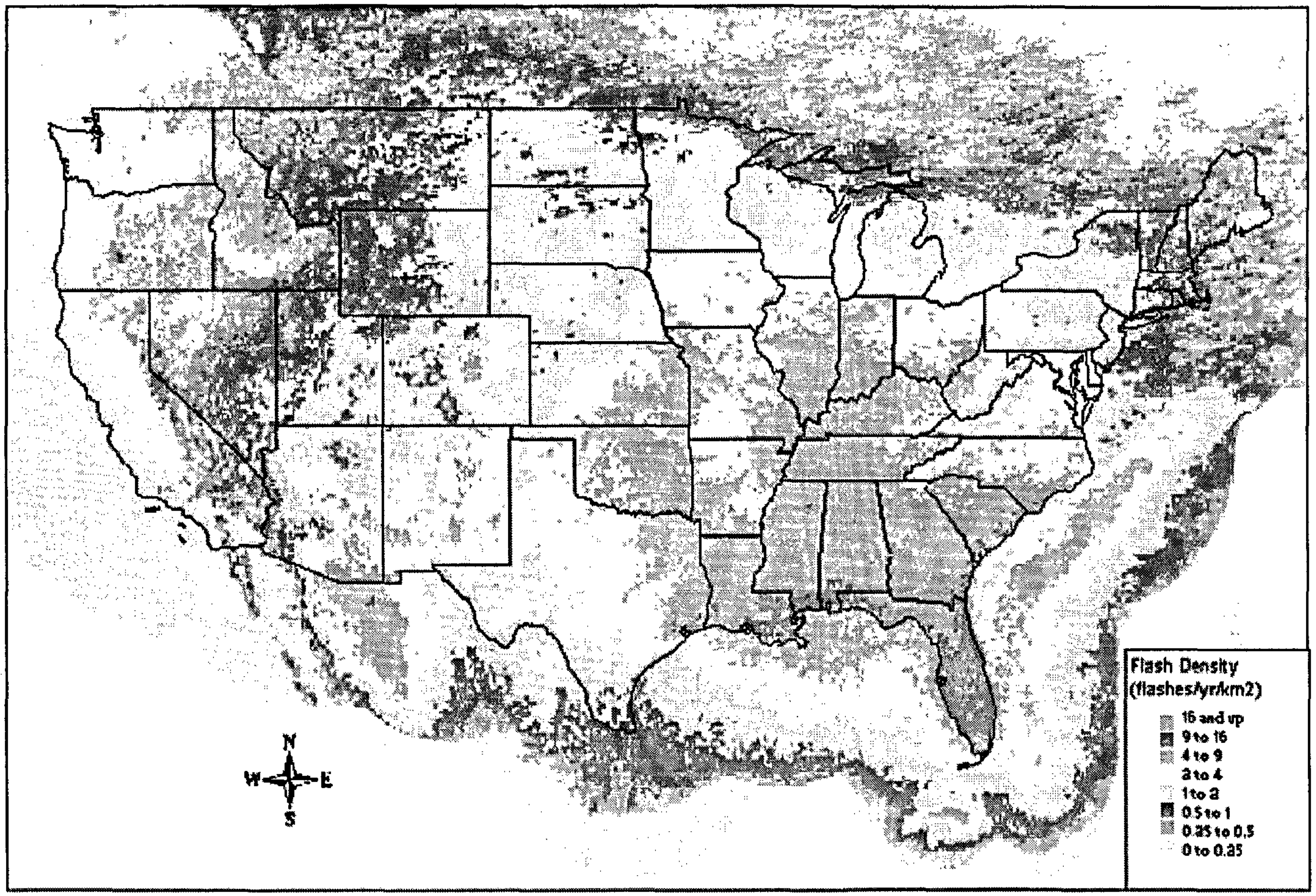
LOCKWOOD GREENE
ENGINEERING & CONSTRUCTION

REFERENCE NUMBER
Figure_3.2-2.dwg

FIGURE 3.2-2
PLOT PLAN
(1 MILE RADIUS)

REVISION 2 DATE: JULY 2004


6019.52'
HIGH POINT OF DITCH
UBC STORAGE PAD
32
SITE BOUNDARY FENCE (TYP.)
3176.73'
EAST DIVERSION DITCH WITH 3:1 SIDE SLOPES
CONTROLLED ACCESS AREA BOUNDARY
FF ELEV. 3415
UBC STORAGE PAD STORMWATER RETENTION BASIN
33
2769.78'
TREATED EFFLUENT EVAPORATIVE BASIN
SITE STORMWATER DETENTION BASIN
5377.12'
NM HIGHWAY 234
NATURAL GAS LINE
OUTFALL OF DIVERSION DITCH
2840.43'
ANDREWS, TX.
CO2 GAS LINE
NEW MEXICO
TEXAS
NORTH
BOUNDA
LEGEND:
1 VISITOR
2 SECURITY
3 ADMINIST
4 LIQUID N
5 CASCADE
6 CASCADE
7 CASCADE
8 ISO FREI
9 UBC STO
10 UF6 HAN
11 FIRE WAT
12 ELECTRIC
13 COOLING
14 CAB
15 CUB
16 TSB
17 CRDB
18 EMPLOYE
19 TRAILER
20 METEORL
21 BLENDING
22 GUARD H



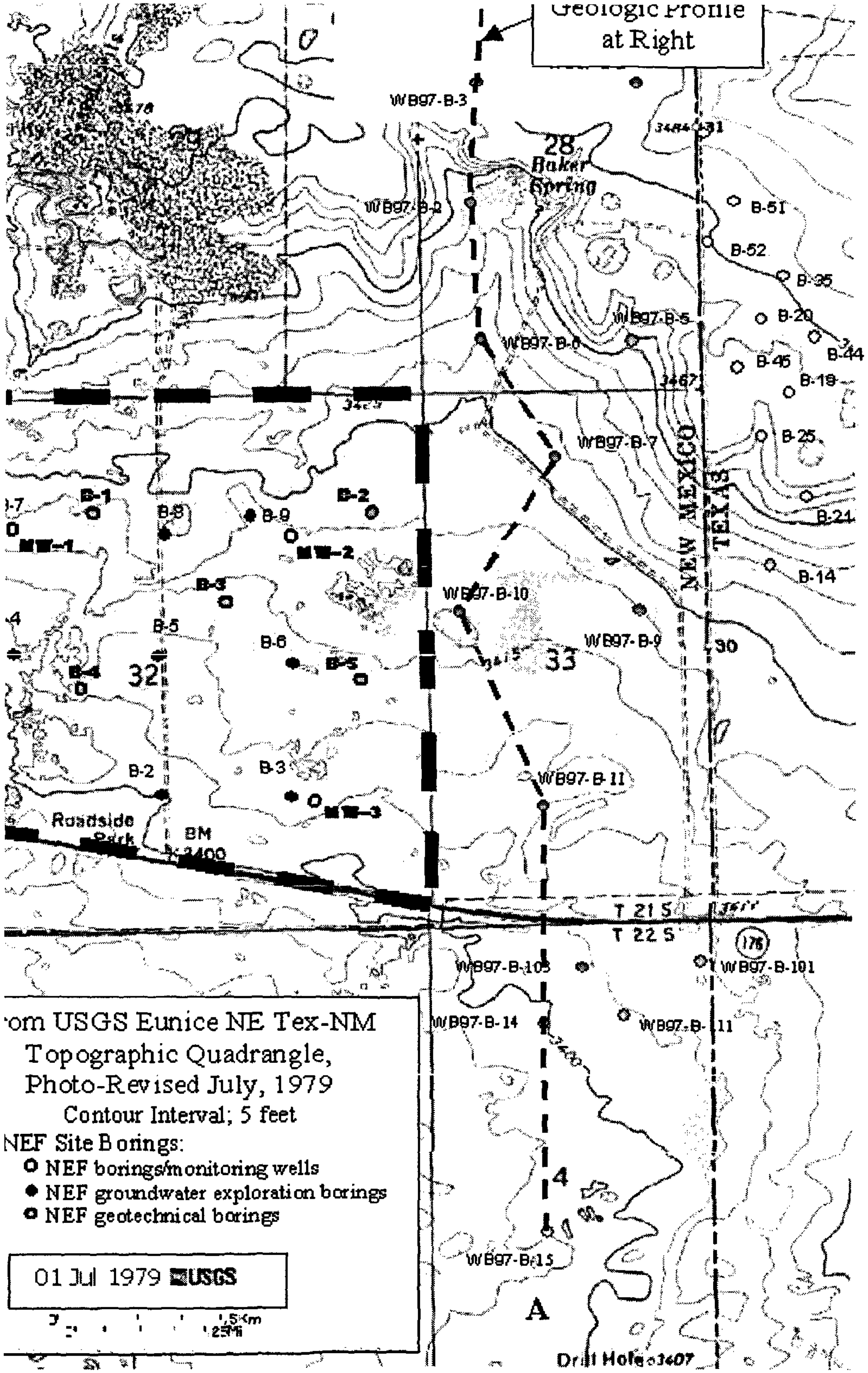
1996-2000 FLASH DENSITY MAP

(NWS, 2003)

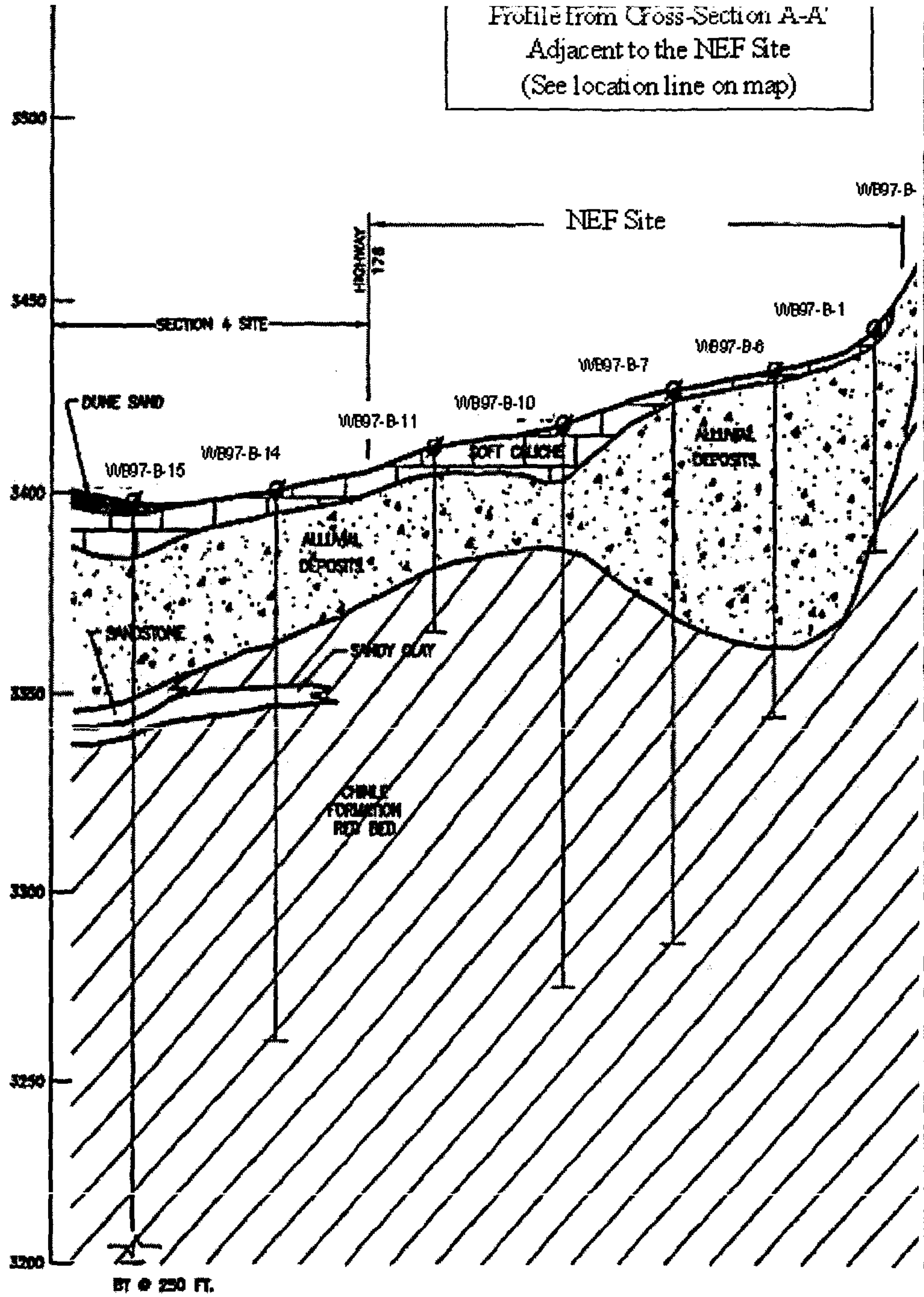
REFERENCE NUMBER
Figure 3.2-4.dwg

FIGURE 3.2-4
AVERAGE LIGHTNING FLASH DENSITY

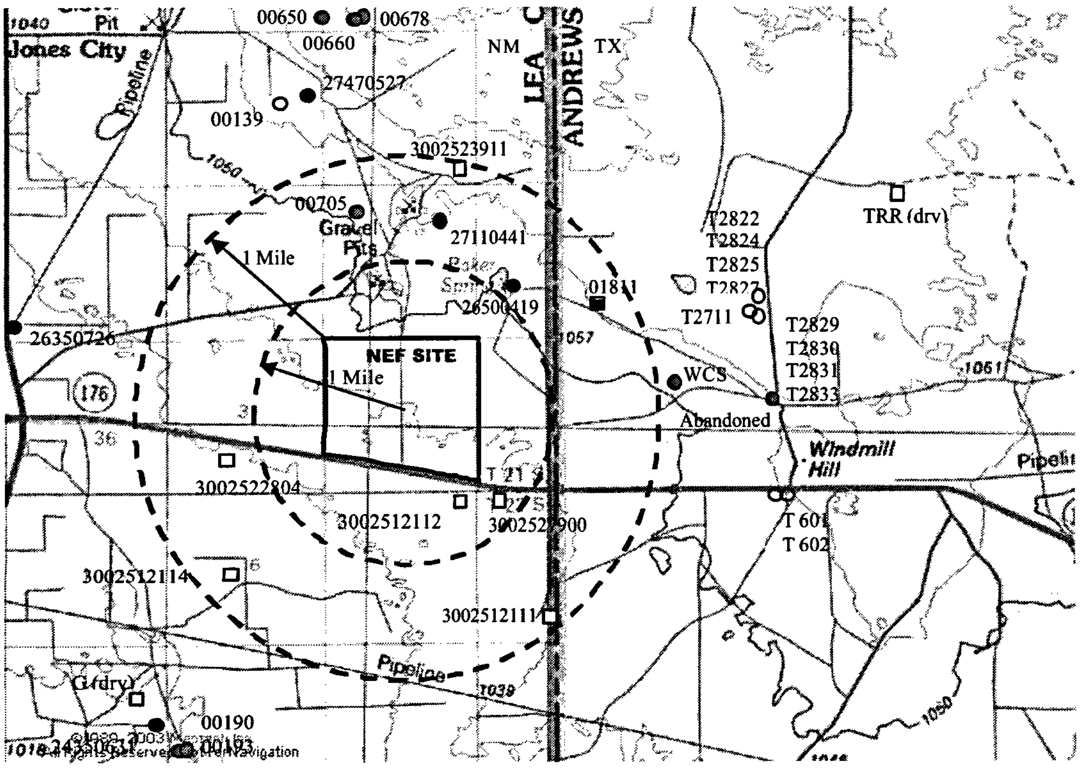
REVISION DATE: DECEMBER 2003



Geologic Profile
at Right
WB97-B-3
28
Baker
Spring
WB97-B-2
B-51
B-52
B-35
WB97-B-5
B-20
WB97-B-6
B-44
B-45
B-19
WB97-B-7
NEW MEXICO
TEXAS
B-25
B-1
B-8
B-9
B-2
B-21
MW-1
MW-2
B-3
B-14
WB97-B-10
B-5
B-6
WB97-B-9
30
B-4
32
B-5
33
B-2
B-3
WB97-B-11
MW-3
Roadside
Park
BM
3400
T 21 S
T 22 S
176
WB97-B-103
WB97-B-101
WB97-B-14
WB97-B-111
4
WB97-B-15
A
Drill Hole 3407
rom USGS Eunice NE Tex-NM
Topographic Quadrangle,
Photo-Revised July, 1979
Contour Interval; 5 feet
NEF Site Borings:
NEF borings/monitoring wells
NEF groundwater exploration borings
NEF geotechnical borings
01 Jul 1979 USGS



Profile from Cross-Section A-A'
Adjacent to the NEF Site
(See location line on map)
3500
3450
3400
3350
3300
3250
3200
NEF Site
HIGHWAY 176
SECTION 4 SITE
WB97-B-1
WB97-B-6
WB97-B-7
WB97-B-10
WB97-B-11
WB97-B-14
WB97-B-15
DUNE SAND
SOFT CALICHE
ALLUVIAL DEPOSITS
ALLUVIAL DEPOSITS
SANDSTONE
SANDY CLAY
CHINLE FORMATION RED BED
BT @ 250 FT.



Water Wells

- Undetermined
- Observation wells: USGS
- Livestock well
- Domestic Well

Oil Wells

- Open symbol represents dry, plugged and/or abandoned hole

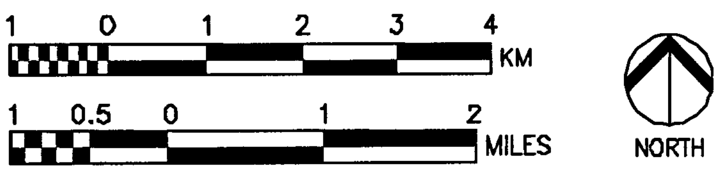


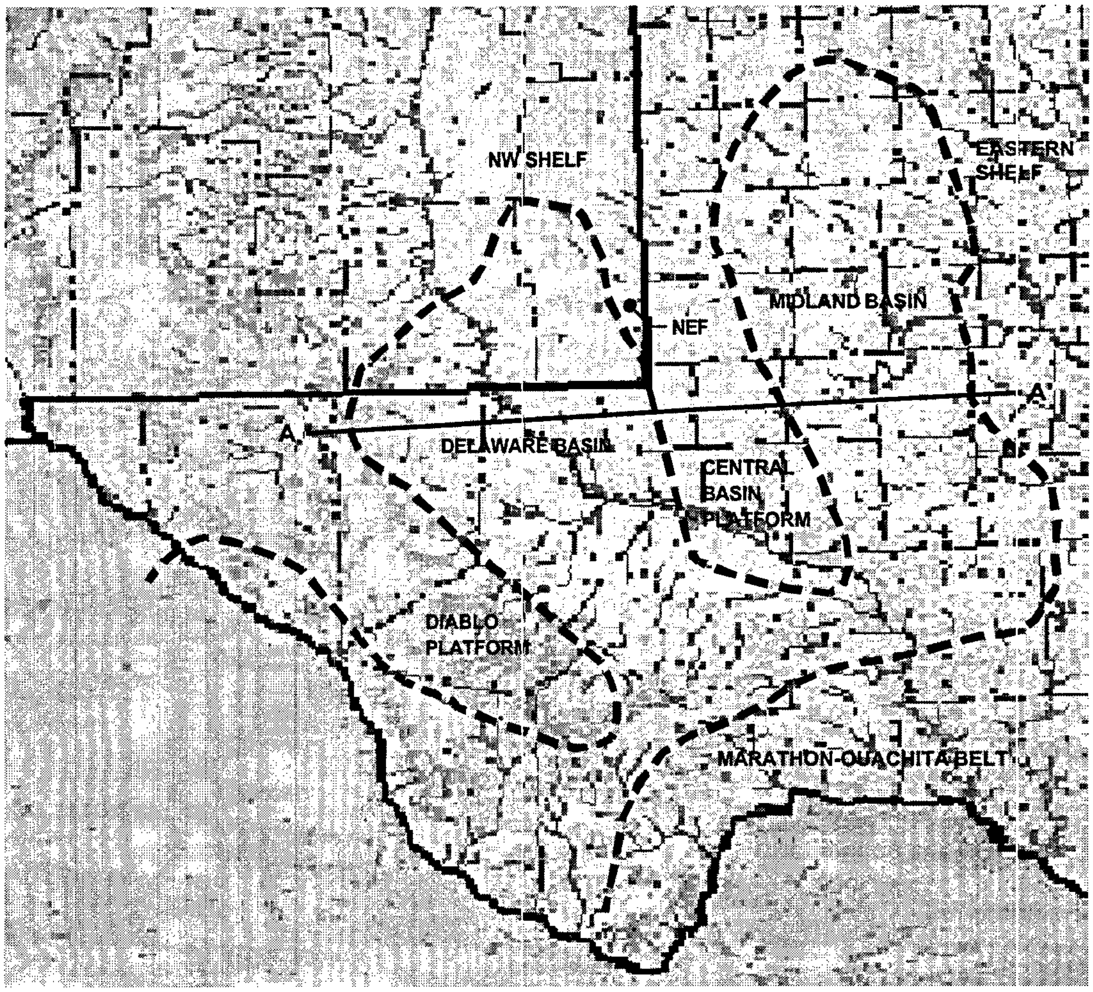
FIGURE 3.2-6

WATER AND OIL WELLS IN THE VICINITY OF THE NEF SITE

REVISION DATE: DECEMBER 2003

REFERENCE NUMBER

Figure 3.2-6.dwg



Permian Basin Geologic Profile

(Generalized from UTPB 2003)

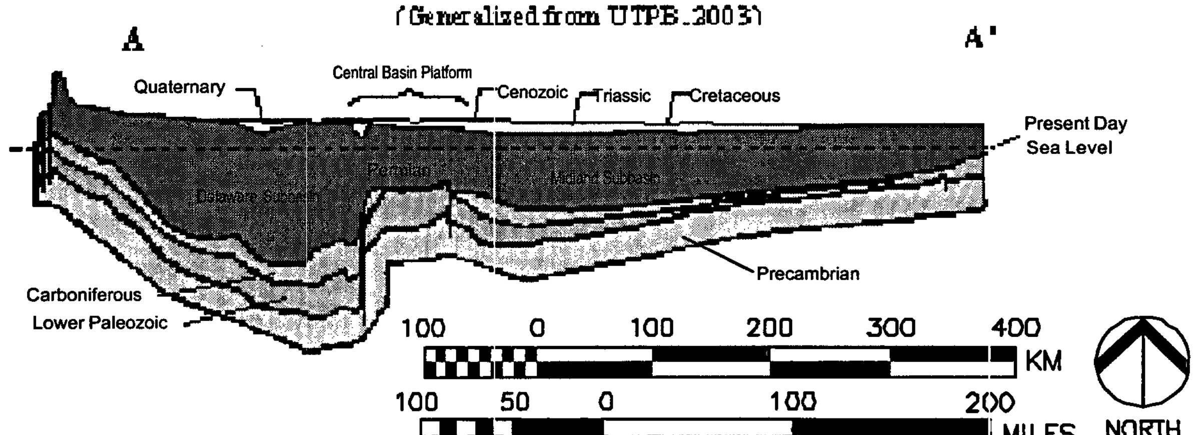
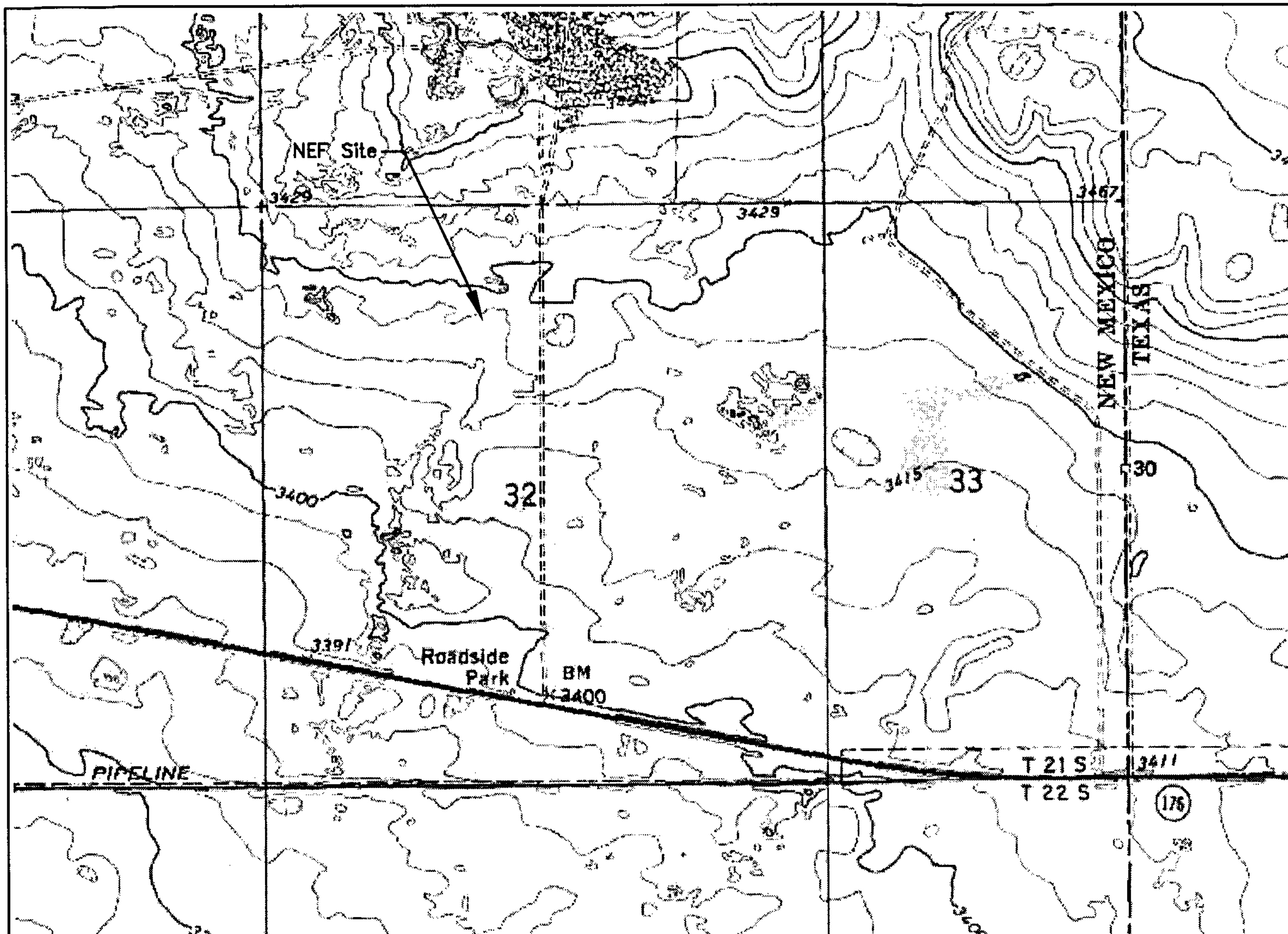


FIGURE 3.2-7

PERMIAN BASIN GEOLOGIC STRUCTURES AND PROFILE

REFERENCE NUMBER
1Figures 3.2.dwg

REVISION DATE: DECEMBER 2003



FROM USGS EUNICE NE TEX-NM TOPOGRAPHIC QUADRANGLE, PHOTO-REVISED JULY, 1979

CONTOUR INTERVAL: 5 FEET

NORTH

0.25 0 0.25 0.5 0.75 1 MILES

Modified from: http://mrdata.usgs.gov.

REFERENCE NUMBER
Figure 3.2-8.dwg

FIGURE 3.2-8

SITE TOPOGRAPHY

REVISION DATE: DECEMBER 2003

;at Plains parts of the state. Thick caliches (locally >20 ft) associ- ssected High Plains surfaces of the Great Plains commonly comprise ence of several carbonate-cemented zones interlayered with reddish l horizons over a basal caprock zone developed on Ogallala (To) ms on various types of parent formations, indicated by subscripts. caliche along Rio Salado northwest of Socorro is partly a travertine buried by sand, the caliche is identified by subscript ca. A distinct- daries are well defined where the caliche forms rimrock and approx- posed in deflation hollows. Where thick and well indurated, caliche r road metal and other aggregate, subject to minimal erosion

FLOODPLAIN AND CHANNEL DEPOSITS ALONG GENERALLY DRY ARROYOS AND WASHES — *Includes deposits along some ntain streams. Extent exaggerated to emphasize drainage patterns. l$_1$, gradients 5 to 15 percent. Arroyos 10 ft deep common. Surface osit was formed by stream overflowing its banks; hummocky where scing fans at mouths of tributaries that crowd the main stream bank; or V-shaped where alluvium grades laterally into fan sand adjoining hillsides. Ephemeral perched water tables under some h of deposits represented has been exaggerated but total area it right because small deposits had to be omitted*

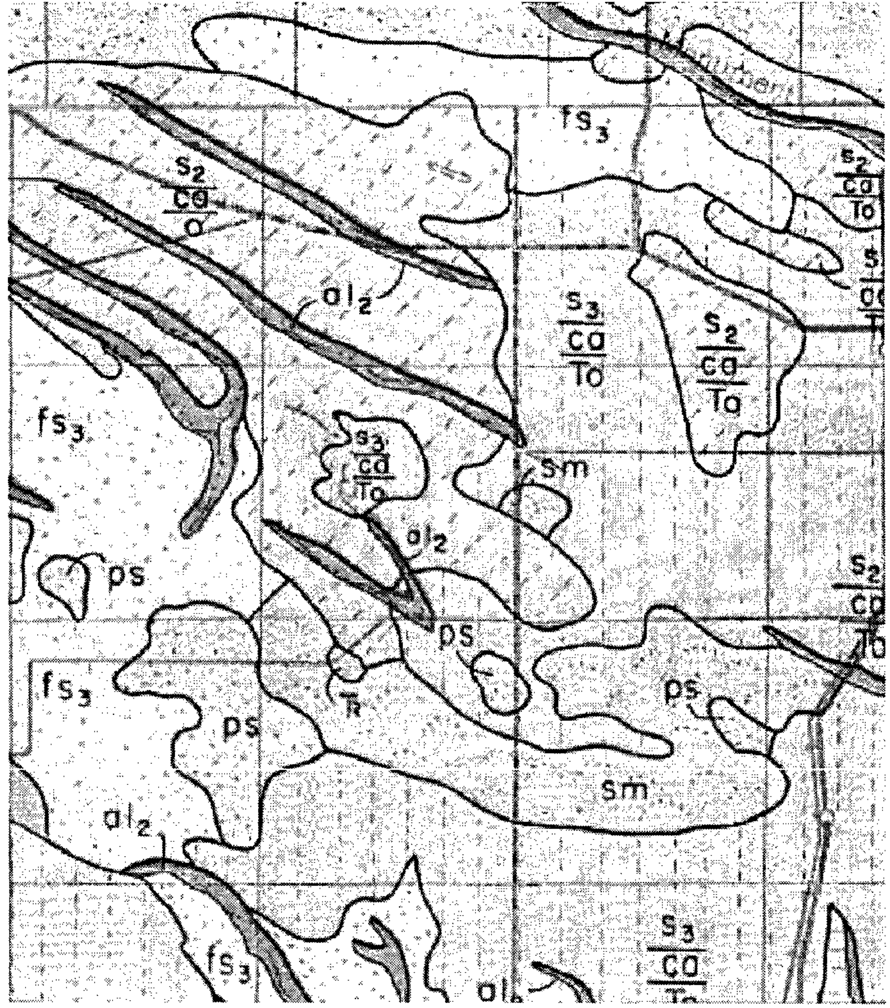
SAND FACIES — *Sandy alluvium with subordinate amounts of fine gravel, silt, and clay. Forms at least four kinds of ground: 1) On ans sloping from the mountains of granitic or gneissic rock (e.g., lorida Mountains), this facies may form a smooth sandy layer a few vering gravel below; slopes 5 to 20 percent; washes 1 to 10 ft ose underlying gravel. 2) On other short fans, sand facies may form toe of fan with slopes averaging 10 percent, commonly reworked dunes 3 to 7 ft high* (sm). *3) Other belts of smooth sandy ground pe 5 percent or less and consist of sand mounds approximately 1 ft che* (fs$_2$). *4) Gypsiferous sand* (fs$_3$), *especially in the Jornada del osa Valley and east side of the Pecos Valley. Sand facies absent on Palomas surface. Thin fan sand covering pediments is denoted by* fs *that identifies underlying formation. Boundary with residual sand, l fan silt is approximate*

MODERATELY THICK SAND ON CALICHE ON OGALLALA FORMATION — *Sand 1 to 3 ft thick. Surface layers noncalcar- lish loam. Local sand mounds. Ground favored for farming. Bound- ate*

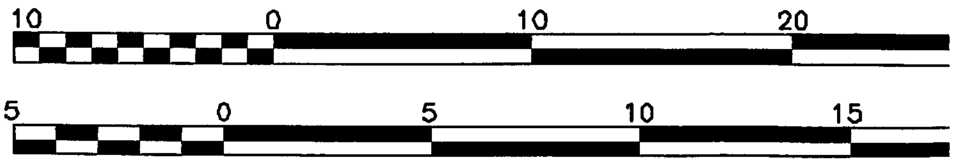
THICK SAND ON CALICHE ON OGALLALA FORMATION *and 3 to 5 ft thick. Local mounds. Brownish-red, fine sandy ddish-brown, sandy clay loam; noncalcareous to depths of us subsoil contains filaments of lime carbonate. Where farmed, ject to wind erosion. Boundaries approximate*

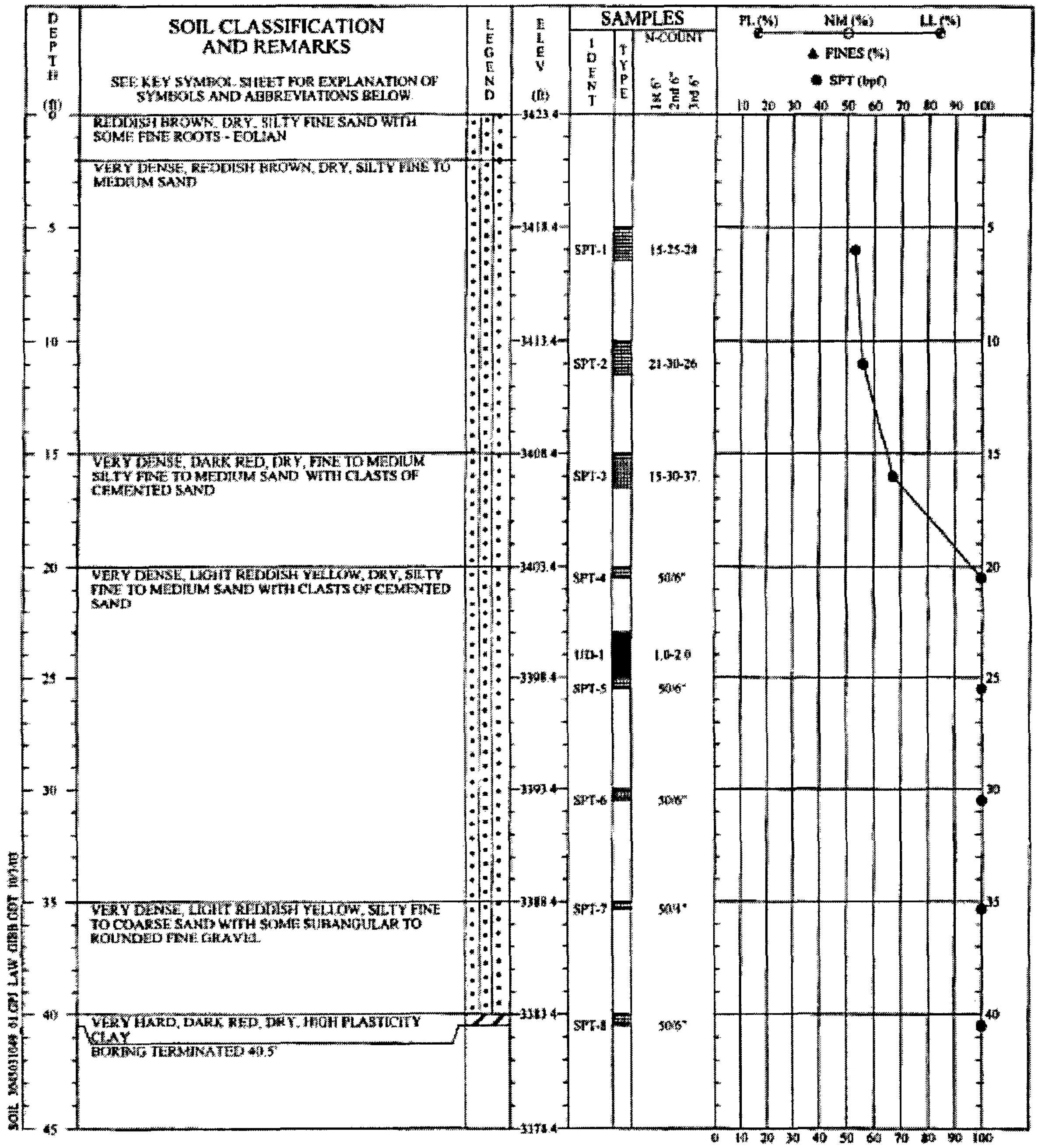
LOOSE SAND IN MOUNDS — *Coppice dunes, commonly to 7 ft high and 25 to 50 ft in diameter; generally elongated but a local exception lies east of Columbus where elongation is Age is Holocene. Boundaries fairly accurate*

ANDY LAKE OR PLAYA DEPOSITS — *Gypsiferous deposits beled* ps$_2$



REFERENCE: (NMIMT, 1977)





REMARKS: STANDARD PENETRATION RESISTANCE TESTING PERFORMED USING A SAFETY HAMMER. NO GROUND WATER ENCOUNTERED AT TIME OF EXPLORATION. BACK FILLED ON 9/9/2003

SOIL TEST BORING RECORD

PROJECT: NEF - Lea County, New Mexico

DRILLED: September 9, 2003 **BORING NO.:** B-1

REFERENCE NUMBER
BORINGS.DWG

FIGURE 3.2-10
SOIL TEST BORING RECORD B-1

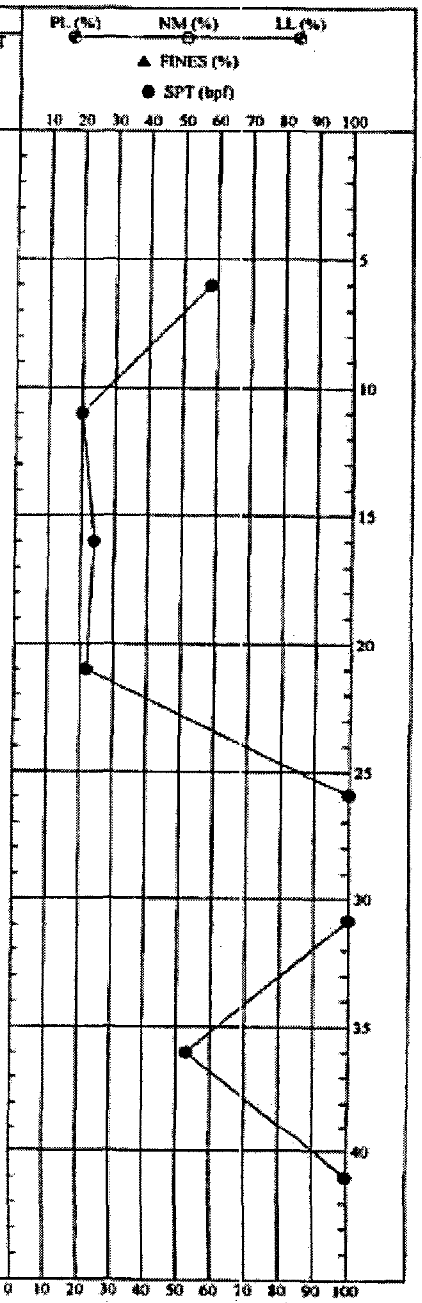
REVISION DATE: DECEMBER 2003

SOIL CLASSIFICATION AND REMARKS

SEE KEY SYMBOL SHEET FOR EXPLANATION OF SYMBOLS AND ABBREVIATIONS BELOW

DEPTH (ft)	SOIL CLASSIFICATION AND REMARKS	ELEV (ft)
0	REDDISH BROWN, DRY, SILTY, FINE SAND WITH SOME FINE ROOTS - EOLIAN	3422.2
	VERY DENSE, LIGHT YELLOW, DRY, FINE SILTY SAND WITH CLASTS OF CEMENTED SAND - CALICHE FROM 6.0' - 7.5'	
5		3417.2
10	FIRM TO VERY FIRM, LIGHT YELLOW, DRY, SILTY FINE TO MEDIUM SAND	3412.2
15		3407.2
20	VERY FIRM TO VERY DENSE, LIGHT YELLOW, SILTY FINE TO MEDIUM SAND WITH CLASTS OF CEMENTED SAND FROM 25.0' TO 35.0'	3402.2
25		3397.2
30		3392.2
35	VERY HARD, DARK RED AND PURPLE, MOIST, HIGH PLASTICITY CLAY	3387.2
40		3382.2
	BORING TERMINATED AT 41.4'	
45		3377.2

SAMPLES IDENT	N-COUNT (1st 6" - 2nd 6" - 3rd 6")
SPT-1	13-29-29
SPT-3	11-10-10
SPT-4	9-12-12
SPT-5	13-10-12
SPT-6	15-50/5"
SPT-7	50/1"
SPT-8	14-23-30
SPT-9	12-33-50/5"



REMARKS: STANDARD PENETRATION RESISTANCE TESTING PERFORMED USING A SAFETY HAMMER. NO GROUND WATER ENCOUNTERED AT TIME OF EXPLORATION. BACK FILLED ON 9/9/2003

SOIL TEST BORING RECORD

PROJECT: NEF - Lea County, New Mexico

DRILLED: September 9, 2003 **BORING NO.:** B-2

SOIL 16453I049 01.GPJ LAW_GIBB.GDT 10/3/03

FIGURE 3.2-11

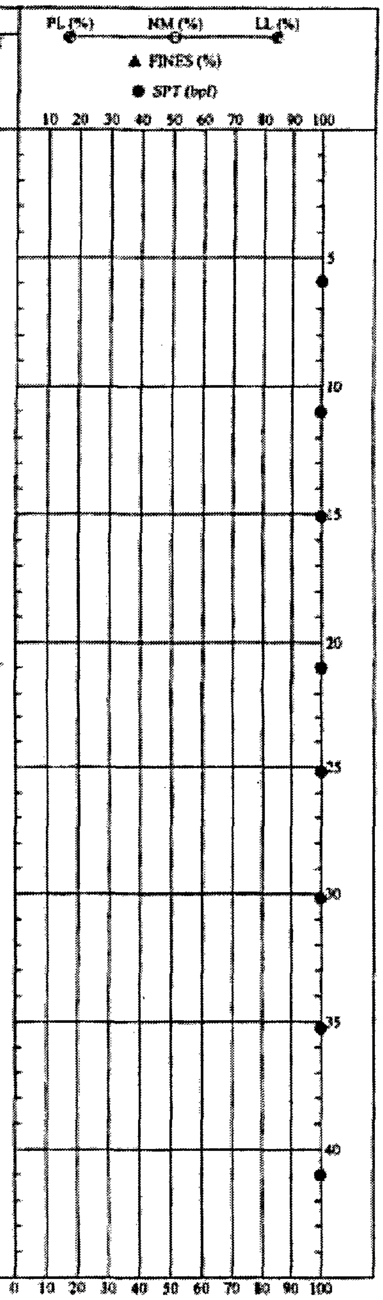
SOIL TEST BORING RECORD B-2

REFERENCE NUMBER BORINGS.DWG

REVISION DATE: DECEMBER 2003

DEPTH (ft)	SOIL CLASSIFICATION AND REMARKS	ELEV (ft)	SAMPLE IDENT	N-COUNT (1st 6", 2nd 6", 3rd 6")
0	REDDISH BROWN, DRY, SILTY FINE SANDY WITH SOME ROOTS - EOLIAN	3412.5		
	VERY DENSE, REDDISH BROWN, DRY, SILTY FINE TO MEDIUM SAND WITH CLASTS OF CEMENTED SAND			
5		3407.5	SPT-1	30-50/5"
10		3402.5	SPT-2	33-50/6"
15	VERY DENSE, LIGHT YELLOW, DRY SILTY FINE TO MEDIUM SAND - CALICHE	3397.5	SPT-3	50/1"
20	VERY DENSE, REDDISH YELLOW, DRY SILTY FINE TO MEDIUM SAND WITH CLASTS OF CEMENTED SAND AND SOME FINE SUBANGULAR TO ROUNDED GRAVEL	3392.5	SPT-4	22-40-50/3"
25	VERY DENSE, LIGHT YELLOW, DRY SILTY FINE TO MEDIUM CEMENTED SAND - CALICHE	3387.5	SPT-5	50/2"
30		3382.5	SPT-6	50/2"
35	VERY DENSE, REDDISH BROWN, DRY, SILTY FINE TO MEDIUM SAND	3377.5	SPT-7	50/3"
40	VERY HARD, DARK RED, DRY, HIGH PLASTICITY CLAY	3372.5	SPT-8	23-50/6"
45		3367.5		

SEE KEY SYMBOL SHEET FOR EXPLANATION OF SYMBOLS AND ABBREVIATIONS BELOW



REMARKS: STANDARD PENETRATION RESISTANCE TESTING PERFORMED USING A SAFETY HAMMER. NO GROUND WATER ENCOUNTERED AT TIME OF EXPLORATION. BACK FILLED ON 9/10/2003

SOIL TEST BORING RECORD

PROJECT: NEF - Lea County, New Mexico

DRILLED: September 10, 2003 **BORING NO.:** B-3

SHEET 1 OF 3

REFERENCE NUMBER
BORINGS.DWG

FIGURE 3.2-12
SOIL TEST BORING RECORD B-3

REVISION DATE: DECEMBER 2003

DEPTH (ft)	SOIL CLASSIFICATION AND REMARKS SEE KEY SYMBOL SHEET FOR EXPLANATION OF SYMBOLS AND ABBREVIATIONS BELOW.	LEGEND	ELEV (ft)	SAMPLES IDENT	TYPE	N-COUNT (1st 6" / 2nd 6" / 3rd 6")
45	VERY HARD, DARK RED, DRY, HIGH PLASTICITY CLAY		3367.5	SPT-9		27-50/5"
50			3362.5	SPT-10		39-50/2"
55	VERY HARD, DARK RED AND PURPLE, DRY, HIGH PLASTICITY CLAY		3357.5	SPT-11		35-50/4"
60			3352.5	SPT-12		31-50/5"
65			3347.5	SPT-13		32-50/4"
70			3342.5	SPT-14		22-50/5"
75			3337.5	SPT-15		39-50/4"
80	VERY HARD, DARK RED AND PURPLE, DRY, HIGH PLASTICITY CLAY WITH OCCASIONAL THIN CEMENTED LAYERS OF GREENISH GRAY		3332.5	SPT-16		50/4"
85	VERY HARD, REDDISH PURPLE WITH GRAY MOTTLING, DRY HIGH PLASTICITY CLAY		3327.5	SPT-17		50/6"
90			3322.5			

PL (%) NM (%) LL (%)

▲ FINES (%)

● SPT (bpf)

0 10 20 30 40 50 60 70 80 90 100

SOIL 3045031049 01.GPJ LAW GINR.GDT 10/3/03

REMARKS STANDARD PENETRATION RESISTANCE TESTING PERFORMED USING A SAFETY HAMMER. NO GROUND WATER ENCOUNTERED AT TIME OF EXPLORATION. BACK FILLED ON 9/10/2003

SOIL TEST BORING RECORD

PROJECT: NEF - Lea County, New Mexico

DRILLED: September 10, 2003 **BORING NO.:** B-3

SHEET 2 OF 3

REFERENCE NUMBER
BORINGS.DWG

FIGURE 3.2-12
SOIL TEST BORING RECORD B-3

REVISION DATE: DECEMBER 2003

DEPTH (ft)	SOIL CLASSIFICATION AND REMARKS (SEE KEY SYMBOL SHEET FOR EXPLANATION OF SYMBOLS AND ABBREVIATIONS BELOW.)	ELEV (ft)	SAMPLES IDENT	N-COUNT	SPT (bpf)
90	VERY HARD, RED AND PURPLE, DRY, HIGH PLASTICITY CLAY	3322.5	SPT-18	50/5"	100
95		3317.5	SPT-19	50/6"	100
100	BORING TERMINATED AT 100.5'	3312.5	SPT-20	50/6"	100
105		3307.5			
110		3302.5			
115		3297.5			
120		3292.5			
125		3287.5			
130		3282.5			
135		3277.5			

PL (%) NM (%) LL (%)
▲ FINES (%)
● SPT (bpf)

SOIL 3049310049 01.GPJ LAW GIBB.GDT 10/3/03

REMARKS: STANDARD PENETRATION RESISTANCE TESTING PERFORMED USING A SAFETY HAMMER. NO GROUND WATER ENCOUNTERED AT TIME OF EXPLORATION. BACK FILLED ON 9/10/2003.

SOIL TEST BORING RECORD

PROJECT: NEF - Lea County, New Mexico

DRILLED: September 10, 2003 **BORING NO.:** B-3

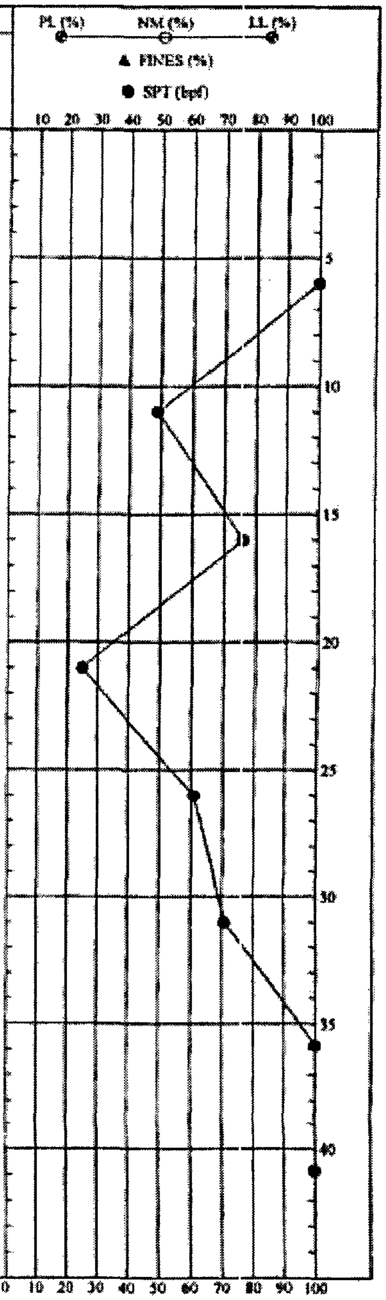
SHEET 3 OF 3

REFERENCE NUMBER
BORINGS.DWG

FIGURE 3.2-12
SOIL TEST BORING RECORD B-3

REVISION DATE: DECEMBER 2003

DEPTH (ft)	SOIL CLASSIFICATION AND REMARKS (SEE KEY SYMBOL SHEET FOR EXPLANATION OF SYMBOLS AND ABBREVIATIONS BELOW)	ELEV (ft)	SAMPLES IDENT	N-COUNT (1st 6", 2nd 6", 3rd 6")
0	REDDISH BROWN, DRY, SILTY FINE SAND WITH SOME ROOTS - EOLIAN	3403.5		
	VERY DENSE, REDDISH BROWN, DRY, SILTY FINE SAND			
5		3398.5	SPT-1	26-50/6"
10	DENSE TO VERY DENSE, LIGHT RED, DRY, SILTY FINE TO MEDIUM SAND WITH CLASTS OF CEMENTED SAND	3393.5	SPT-2	17-26-23
15		3388.5	SPT-3	20-39-37
20	VERY FIRM TO VERY DENSE, LIGHT RED, DRY, SILTY FINE SAND WITH CEMENTED ZONES FROM 25.0' TO 30.0'	3383.5	SPT-4	7-13-12
			UD-2	1.0-2.0
25		3378.5	SPT-5	16-34-27
30	VERY DENSE, LIGHT REDDISH BROWN, DRY FINE SILTY FINE TO MEDIUM SAND	3373.5	SPT-6	16-30-41
35		3368.5	SPT-7	39-50/4"
40	VERY HARD, DARK RED, DRY, HIGH PLASTICITY CLAY	3363.5	SPT-8	35-50/4"
	BORING TERMINATED AT 40 ft			
45		3358.5		



SOIL_3045011... GPJ LAW GIBB GDT 10/2/03

REMARKS: STANDARD PENETRATION RESISTANCE TESTING PERFORMED USING A SAFETY HAMMER. NO GROUND WATER ENCOUNTERED AT TIME OF EXPLORATION. BACK FILLED ON 9/9/2003.

SOIL TEST BORING RECORD

PROJECT: NEF - Lea County, New Mexico

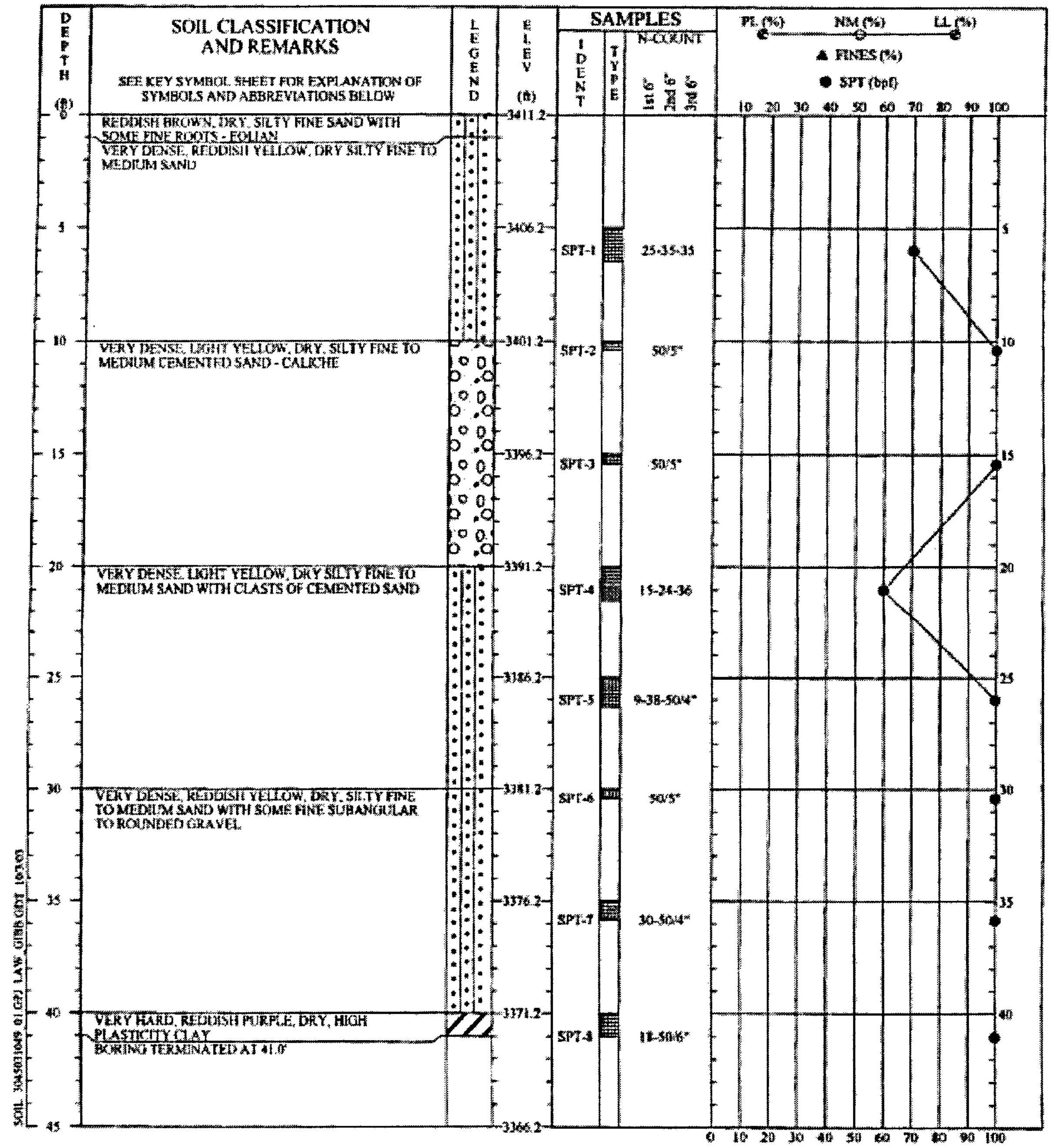
DRILLED: September 9, 2003 **BORING NO.:** B-4

FIGURE 3.2-13

SOIL TEST BORING RECORD B-4

REFERENCE NUMBER
BORINGS.DWG

REVISION DATE: DECEMBER 2003



REMARKS STANDARD PENETRATION RESISTANCE TESTING PERFORMED USING A SAFETY HAMMER. NO GROUND WATER ENCOUNTERED AT TIME OF EXPLORATION. BACK FILLED ON 9/10/2003

SOIL TEST BORING RECORD

PROJECT: NEF - Lea County, New Mexico

DRILLED: September 10, 2003 **BORING NO.:** B-5

FIGURE 3.2-14

SOIL TEST BORING RECORD B-5

REFERENCE NUMBER
BORINGS.DWG

REVISION DATE: DECEMBER 2003

GROUP SYMBOLS	TYPICAL NAMES	GROUP SYMBOLS	TYPICAL NAMES
	TOPSOIL		CONCRETE
	ASPHALT		DOLOMITE
	GRAVEL		LIMESTONE
	FILL		SHALE
	SUBSOIL		LIMESTONE/SHALE - Limestone with shale interbeds
	ALLUVIUM		SANDSTONE
	COLLUVIUM		SILTSTONE
	RESIDUUM - Soft to firm		AUGER BORING
	RESIDUUM - Stiff to very hard		UNDISTURBED SAMPLE ATTEMPT

BOUNDARY CLASSIFICATIONS: Soils possessing characteristics of two groups are designated by combinations of group symbols.

SILT OR CLAY	SAND			GRAVEL		Cobbles	Boulders
	Fine	Medium	Coarse	Fine	Coarse		

No.200 No.40 No.10 No.4 3/4" 3" 12"

U.S. STANDARD SIEVE SIZE

Reference: The Unified Soil Classification System, Corps of Engineers, U.S. Army Technical Memorandum No. 3-357, Vol. 1, March, 1953 (Revised April, 1960)

	Undisturbed Sample 1.5-2.0 = Recovered (ft) / Pushed (ft)		
	Split Spoon Sample		Auger Cuttings
	Rock Core 60-100 = RQD / Recovery		Dilatometer
	No Sample		Crandall Sampler
	Rotary Drill		Pressure Meter
	Water Table at time of drilling		No Recovery
			Water Table after 24 hours

Correlation of Penetration Resistance with Relative Density and Consistency

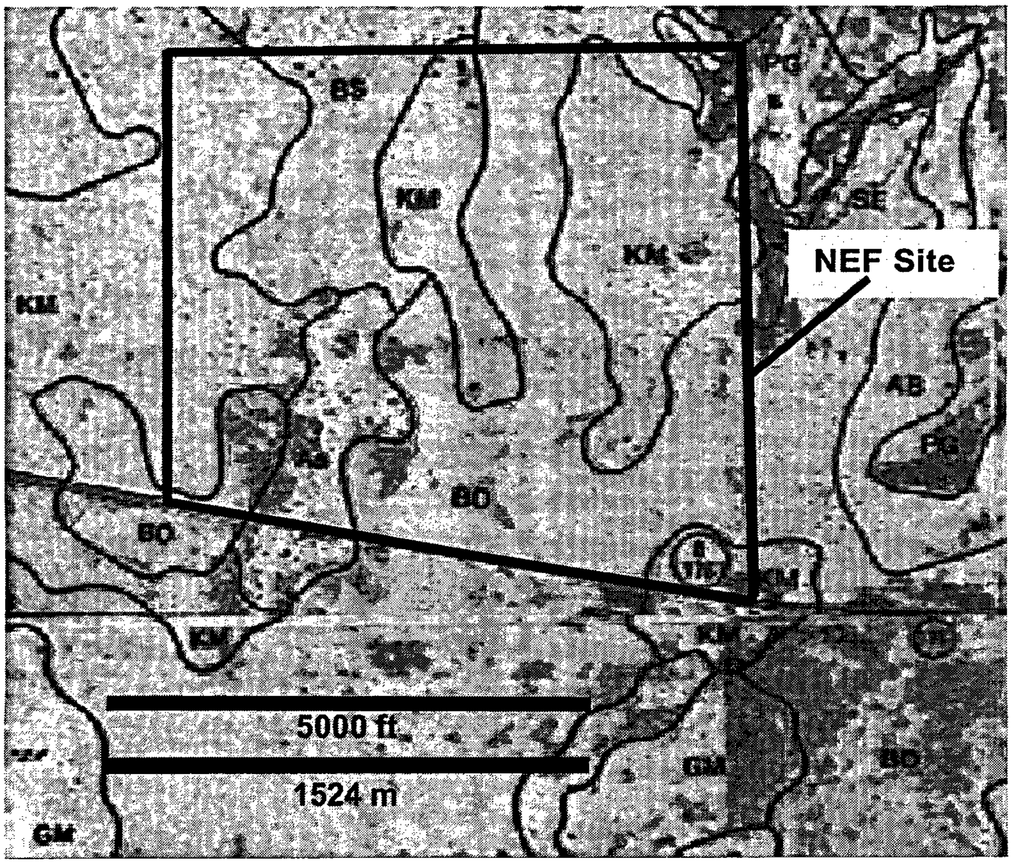
SAND & GRAVEL		SILT & CLAY	
No. of Blows	Relative Density	No. of Blows	Consistency
0 - 4	Very Loose	0 - 2	Very Soft
5 - 10	Loose	3 - 4	Soft
11 - 20	Firm	5 - 8	Firm
21 - 30	Very Firm	9 - 15	Stiff
31 - 50	Dense	16 - 30	Very Stiff
Over 50	Very Dense	31 - 50	Hard
		Over 50	Very Hard

boringkey.dwg
BORINGFIGURES.dwg

FIGURE 3.2-15

SOIL TEST BORING KEY TO SYMBOLS AND DESCRIPTIONS

REVISION DATE: DECEMBER 2003



USDA SOIL DESIGNATION	SOIL NAME/DESCRIPTION	UNIFIED SOIL CLASSIFICATION DESIGNATION(S)
As	ACTIVE (SAND) DUNE LAND.	SP
BO	BROWNFIELD-SPRINGER ASSOCIATION: MOSTLY FINE SAND WITH LOAM FINE SAND; LEVEL TO UNDULATING TOPOGRAPHY; MODERATELY RAPID PERMEABLILITY AND SLOW RUNOFF.	SM
BS	BROWNFIELD-SPRINGER ASSOCIATION: MOSTLY FINE SAND WITH LOAM FINE SAND; DUNES AND HUMMOCKS FOR CONCAVE AND CONVEX ROLLING TERRAIN; DRAINAGE SIMILAR TO BO.	SM
KM	KERMIT SOILS AND DUNE LAND: EXCESSIVELY-DRAINED NON-CALCAREOUS SOILS; HUMMOCKY AND UNDULATING TOPOGRAPHY DUE TO EOLIAN PROCESSES.	SP-SM OR SM
MU	MIXED ALLUVIAL LANDS: UNCONSOLIDATED, STRATIFIED ALLUVIUM WITH VARIED TEXTURES OCCURRING INTERMITTENTLY IN DRAINAGE-WAYS A FEW FEET IN THICKNESS; MODERATE TO RAPID PERMEABILITY WITH SLOW RUNOFF.	VARIABLE
PG	PORTALES AND GOMEZ FINE SANDY LOAMS: LIGHT CLAY LOAM, WELL-DRAINED.	VARIABLE

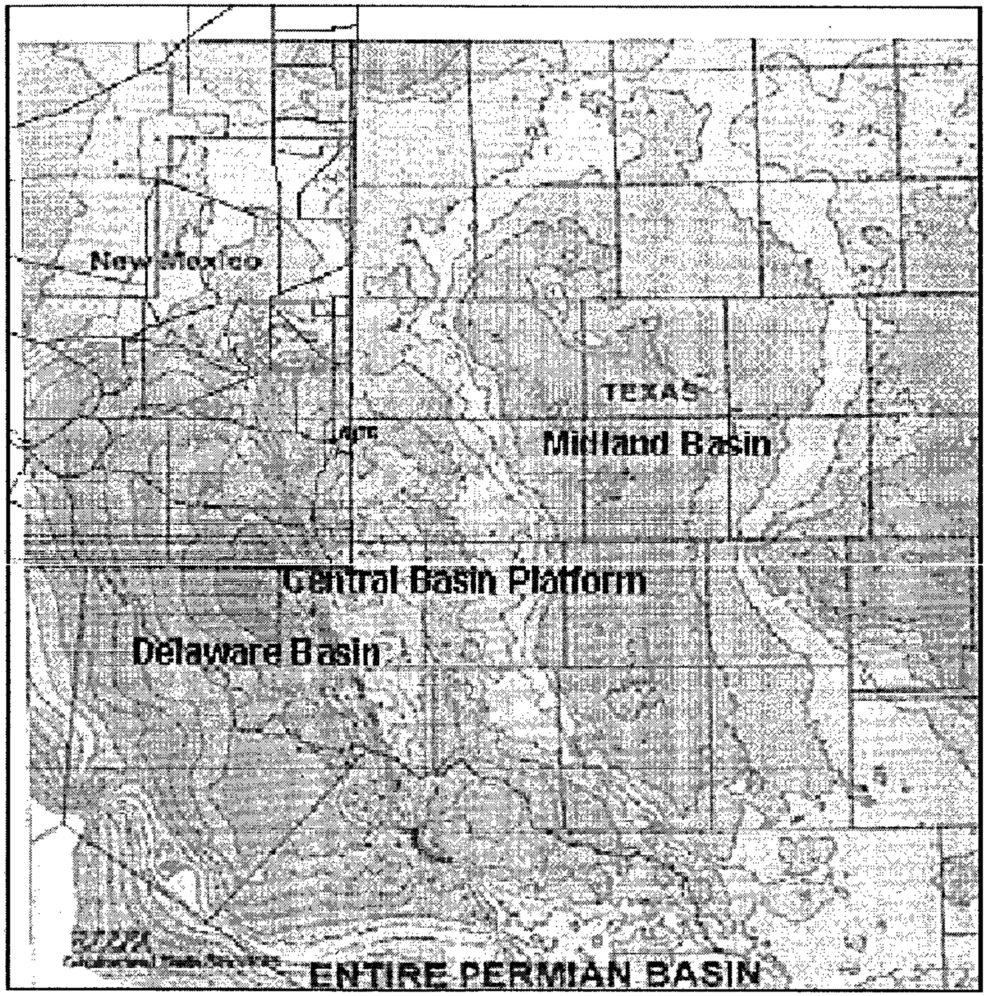
SOURCE: (USDA, 1974)

NORTH

REFERENCE NUMBER
Figure 3.2-16.dwg

FIGURE 3.2-16
SITE SOILS MAP
(USDA, 1974)

REVISION DATE: DECEMBER 2003



LEGEND

☆ NEF SITE

REFERENCE:

TALLEY, D.J., 1997, CHARACTERIZATION OF A SAN ANDRES CARBONATE RESERVOIR USING FOUR DIMENSIONAL, MULTICOMPONENT ATTRIBUTE ANALYSIS, MASTER OF SCIENCE THESIS, COLORADO SCHOOL OF MINES.

NORTH

60 0 60 120 180 KM

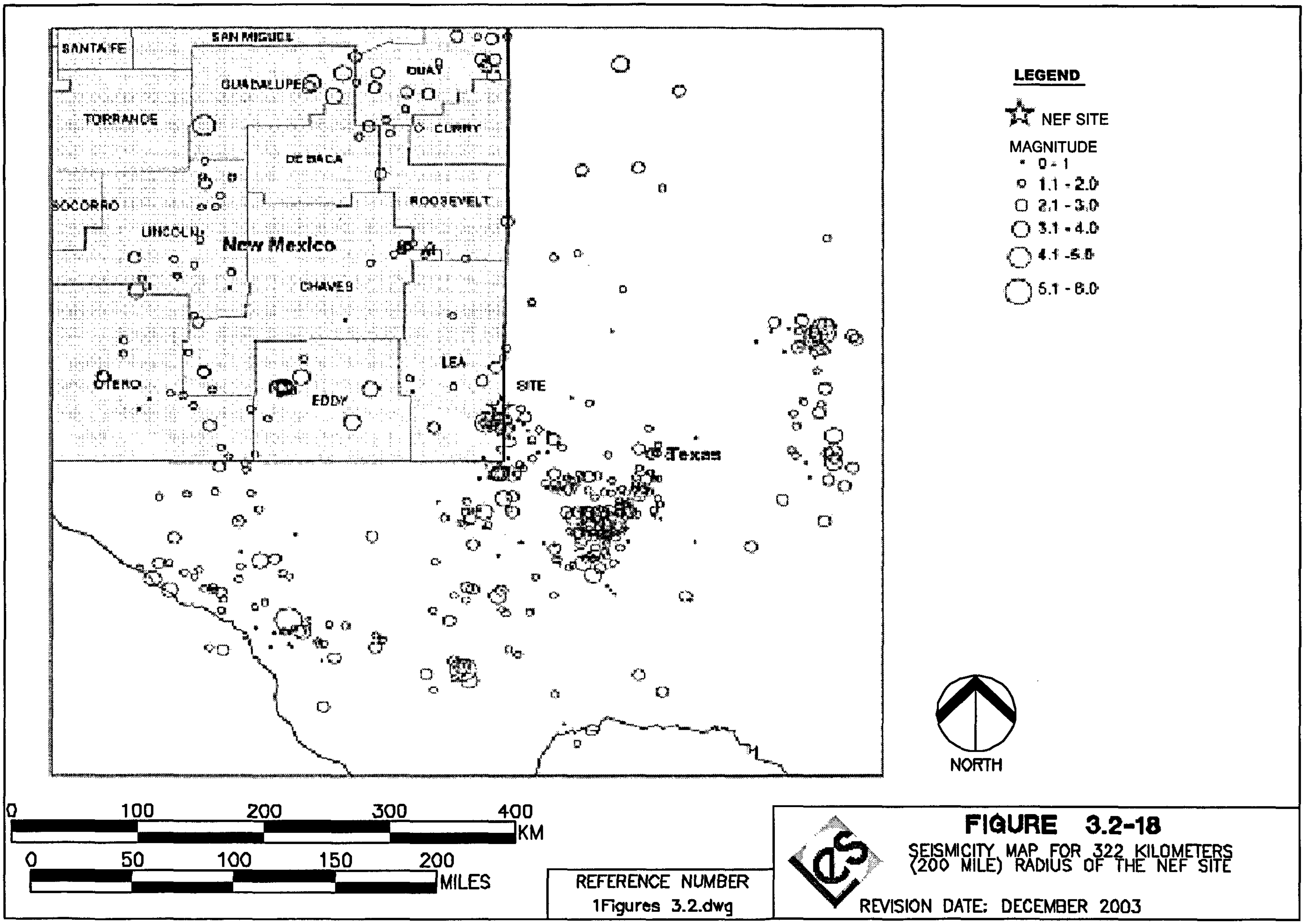
30 0 30 60 90 120 MILES

REFERENCE NUMBER
1Figures 3.2.dwg

FIGURE 3.2-17

TECTONIC SUBDIVISIONS OF THE PERMIAN BASIN

REVISION DATE: DECEMBER 2003

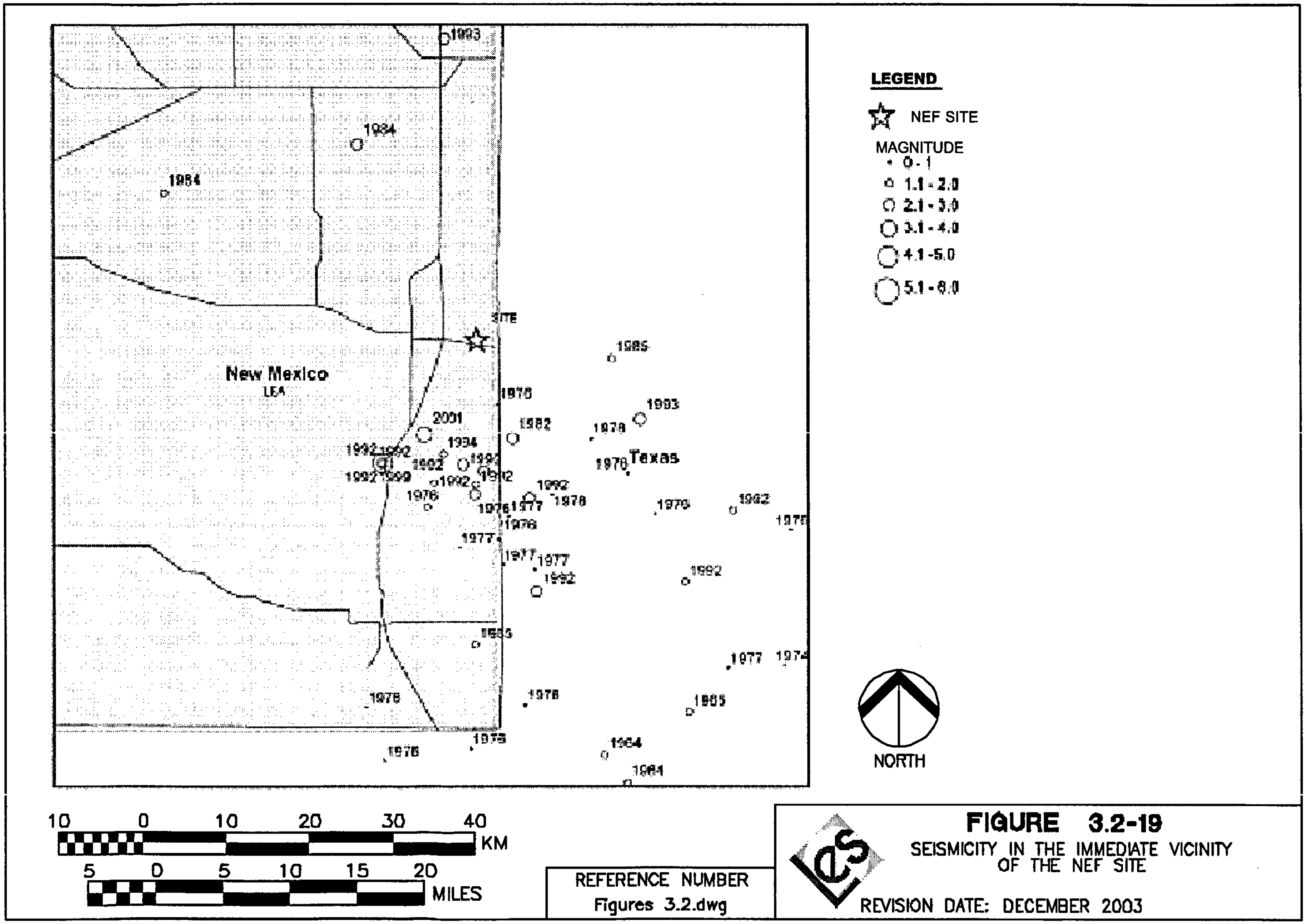


REFERENCE NUMBER
1Figures 3.2.dwg

FIGURE 3.2-18

SEISMICITY MAP FOR 322 KILOMETERS (200 MILE) RADIUS OF THE NEF SITE

REVISION DATE: DECEMBER 2003

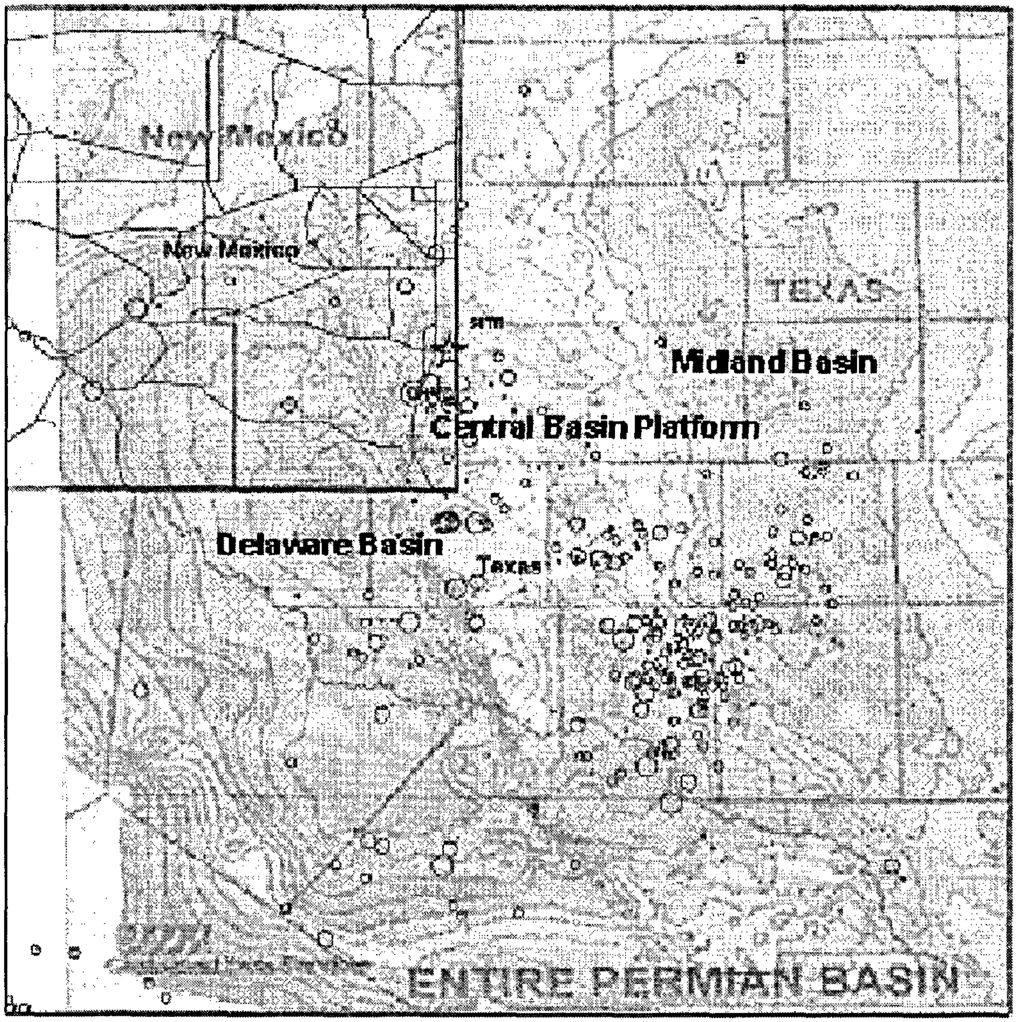


REFERENCE NUMBER
Figures 3.2.dwg

FIGURE 3.2-19

SEISMICITY IN THE IMMEDIATE VICINITY OF THE NEF SITE

REVISION DATE: DECEMBER 2003



LEGEND

☆ NEF SITE

MAGNITUDE

- 0 - 1
- 1.1 - 2.0
- 2.1 - 3.0
- 3.1 - 4.0
- 4.1 - 5.0
- 5.1 - 6.0

NORTH

40 0 40 80 120 160 KM

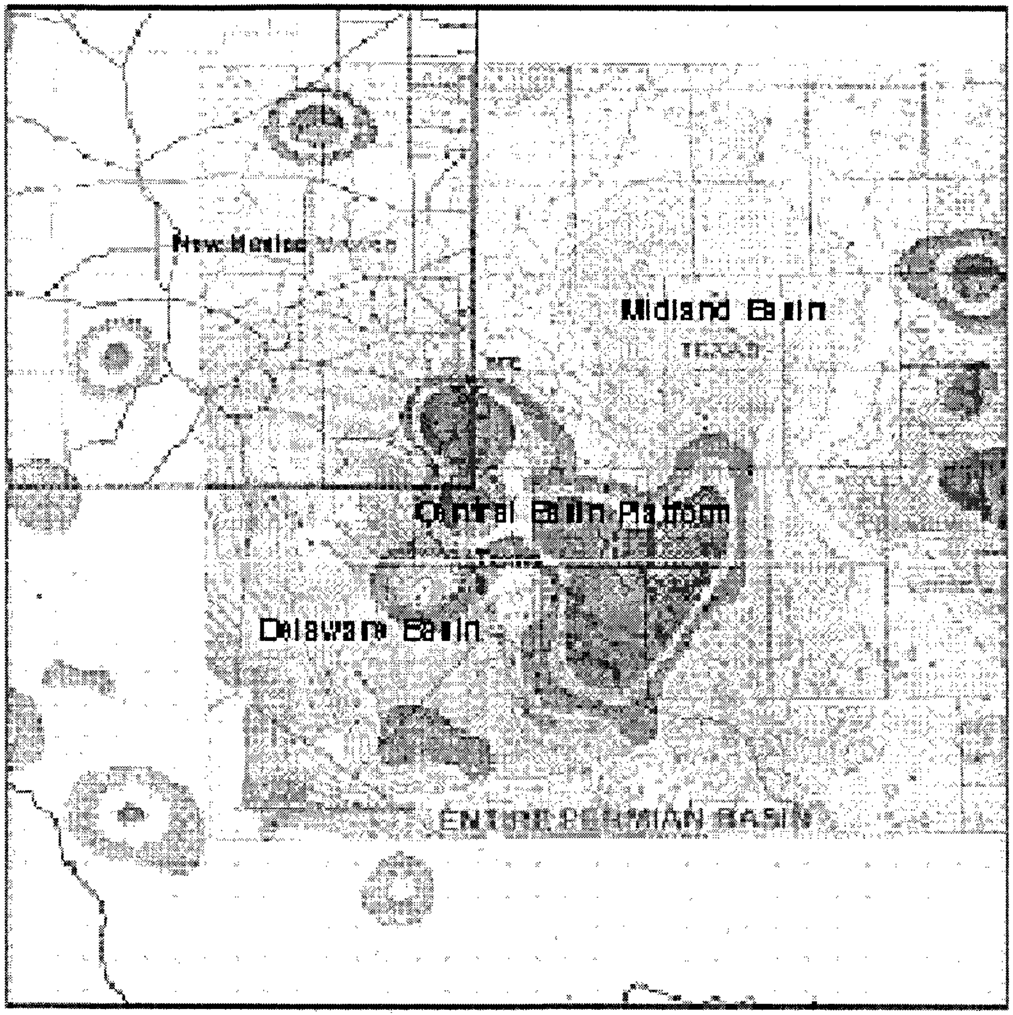
20 0 20 40 60 80 MILES

REFERENCE NUMBER
Figures 3.2.dwg

FIGURE 3.2-20

REGIONAL SEISMICITY AND TECTONIC ELEMENTS OF THE PERMIAN BASIN

REVISION DATE: DECEMBER 2003



LEGEND

NEF SITE

epicenters
EARTHQUAKE DENSITY
[illegible]
[illegible]
[illegible] - 133
133 - 705

NOTE:
THE EARTHQUAKE FREQUENCY CONTOURS SHOWN PROVIDE A VISUAL PORTRAYAL OF THE AREAS WITH SIMILAR EARTHQUAKE COUNTS PER AREA, i.e., EARTHQUAKE DENSITY. THE DENSITY UNITS THEMSELVES ARE NOT ABSOLUTE, BUT A RELATIVE REPRESENTATION OF EARTHQUAKE FREQUENCY FROM ONE LOCATION TO ANOTHER LOCATION. 2

NORTH

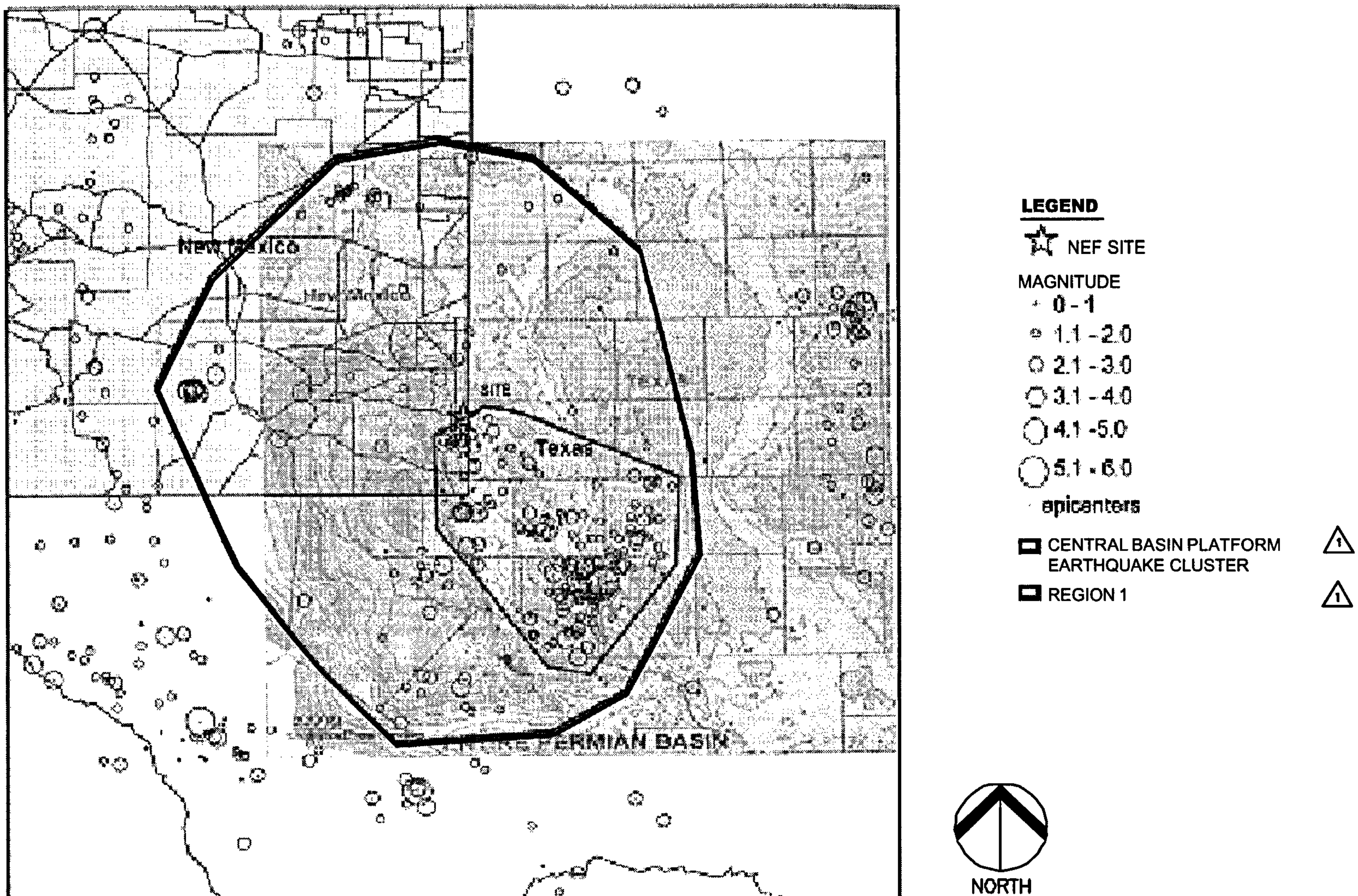
60 0 60 120 180 240 KM

30 0 30 60 90 120 150 MILES

REFERENCE NUMBER
Figures 3.2.dwg

FIGURE 3.2-21
EARTHQUAKE FREQUENCY CONTOURS AND TECTONIC ELEMENTS OF THE PERMIAN BASIN

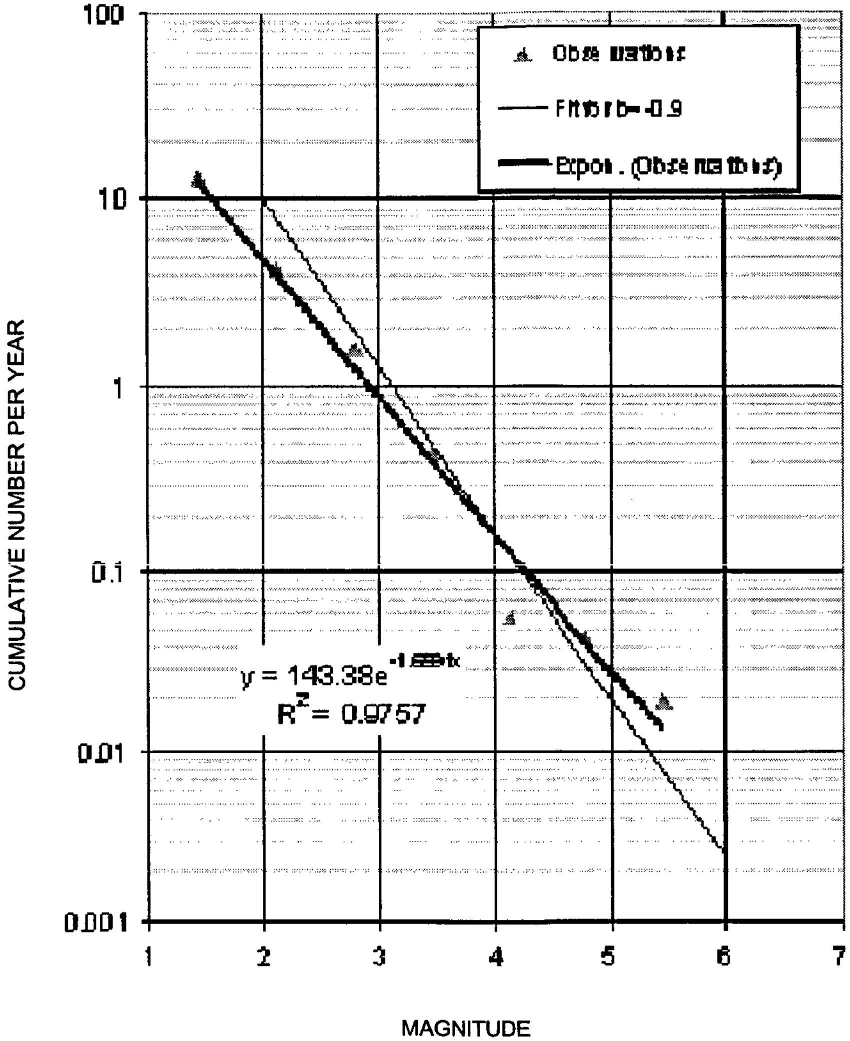
REVISION 2 DATE: JULY 2004



REFERENCE NUMBER
2Figures3.2.dwg

FIGURE 3.2-22
SEISMIC SOURCE AREAS FOR EARTHQUAKE FREQUENCY STATISTICAL ANALYSES

REVISION 1 DATE: FEBRUARY 2004



Exponential Best Fit

$Log(Nc) = 2.15 - 0.74(M)$

Fit assuming b = -0.90

$Log(Nc) = 2.80 - 0.90(M)$

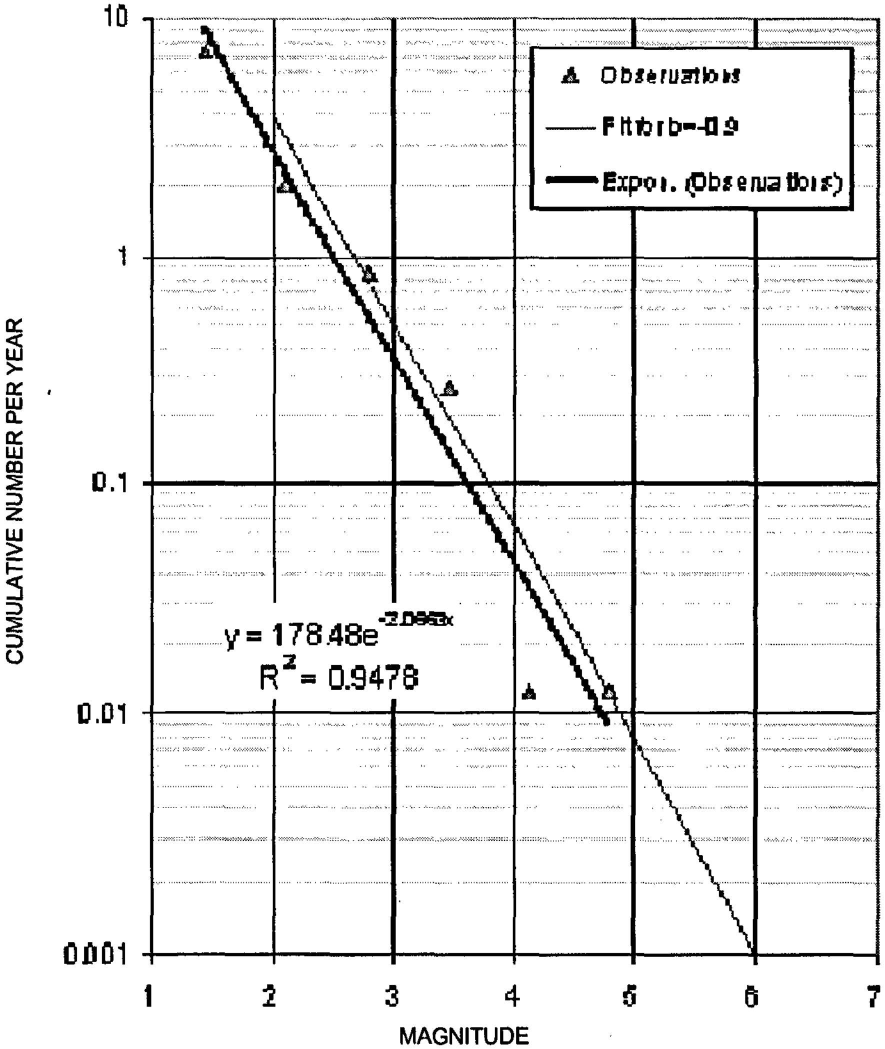
M IS MAGNITUDE SCALED TO DURATION
MAGNITUDE AS DESCRIBED IN
NEW MEXICO TECH CATALOGS
CIRULAR 210, 2002

REFERENCE NUMBER
2Figures3.2.dwg

FIGURE 3.2-23

EARTHQUAKE RECURRENCE MODELS FOR THE 322 KM (200 MILE) RADIUS COMPOSITE CATALOG

REVISION DATE: DECEMBER 2003



Exponential Best Fit

$Log(Nc) = 2.25 - 0.89(M)$

Fit assuming b = -0.90

$Log(Nc) = 2.40 - 0.90(M)$

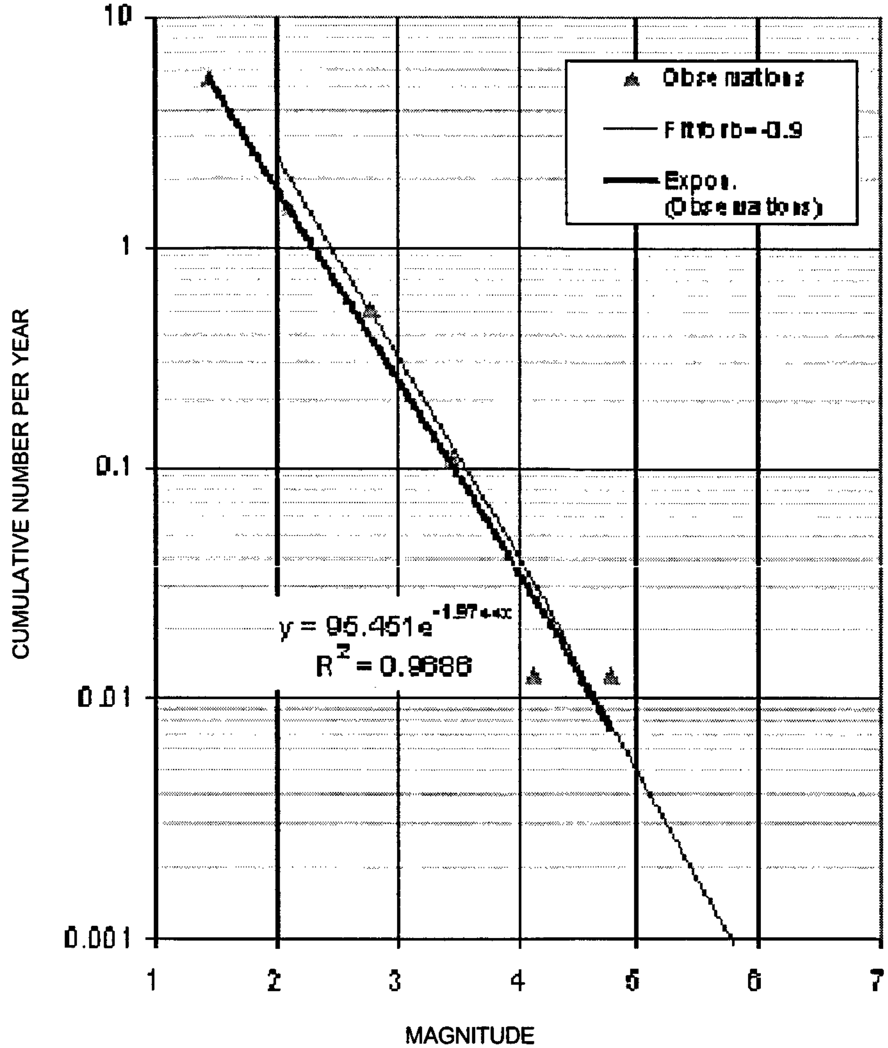
M IS MAGNITUDE SCALED TO DURATION
MAGNITUDE AS DESCRIBED IN
NEW MEXICO TECH CATALOGS
CIRULAR 210, 2002

REFERENCE NUMBER
2Figures3.2.dwg

FIGURE 3.2-24

EARTHQUAKE RECURRENCE MODELS FOR REGION 1
(161 KM (100 MILE) RADIUS OF NEF SITE)

REVISION DATE: DECEMBER 2003



Exponential Best Fit

Log(Nc) = 1.98 – 0.86(M)

Fit assuming b = -0.90

Log(Nc) = 2.20 – 0.90(M)

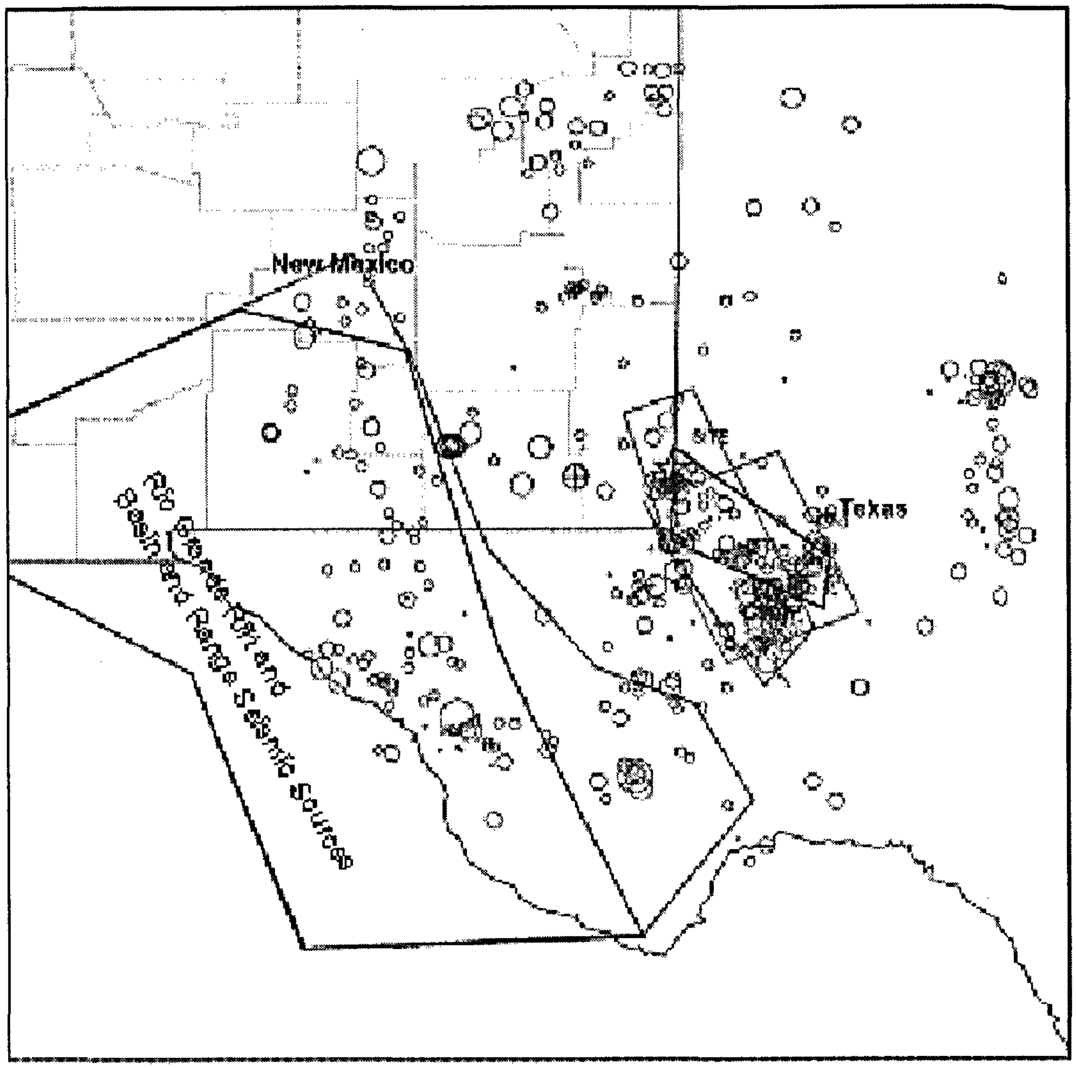
M IS MAGNITUDE SCALED TO DURATION
MAGNITUDE AS DESCRIBED IN
NEW MEXICO TECH CATALOGS
CIRULAR 210, 2002

REFERENCE NUMBER
2Figures3.2.dwg

FIGURE 3.2-25

EARTHQUAKE RECURRENCE MODELS FOR REGION 2
(CBP HIGHER DENSITY EARTHQUAKE CLUSTER)

REVISION DATE: DECEMBER 2003



LEGEND

- NEF SITE
- MAGNITUDE
  - 0 - 1
  - 1.1 - 2.0
  - 2.1 - 3.0
  - 3.1 - 4.0
  - 4.1 - 5.0
  - 5.1 - 6.0
- BASIN AND RANGE Sub A
- BASIN AND RANGE Sub B
- CENTRAL BASIN PLATFORM A
- CENTRAL BASIN PLATFORM B
- CENTRAL BASIN PLATFORM C
- WIPP SITE

REFERENCE:

DOE/WIPP-95-2065 REV. 7
SAFETY ANALYSIS REPORT FOR THE WASTE ISOLATION PILOT PLANT, FIG. 2.5-26

NORTH

100 0 100 200 300 400 KM

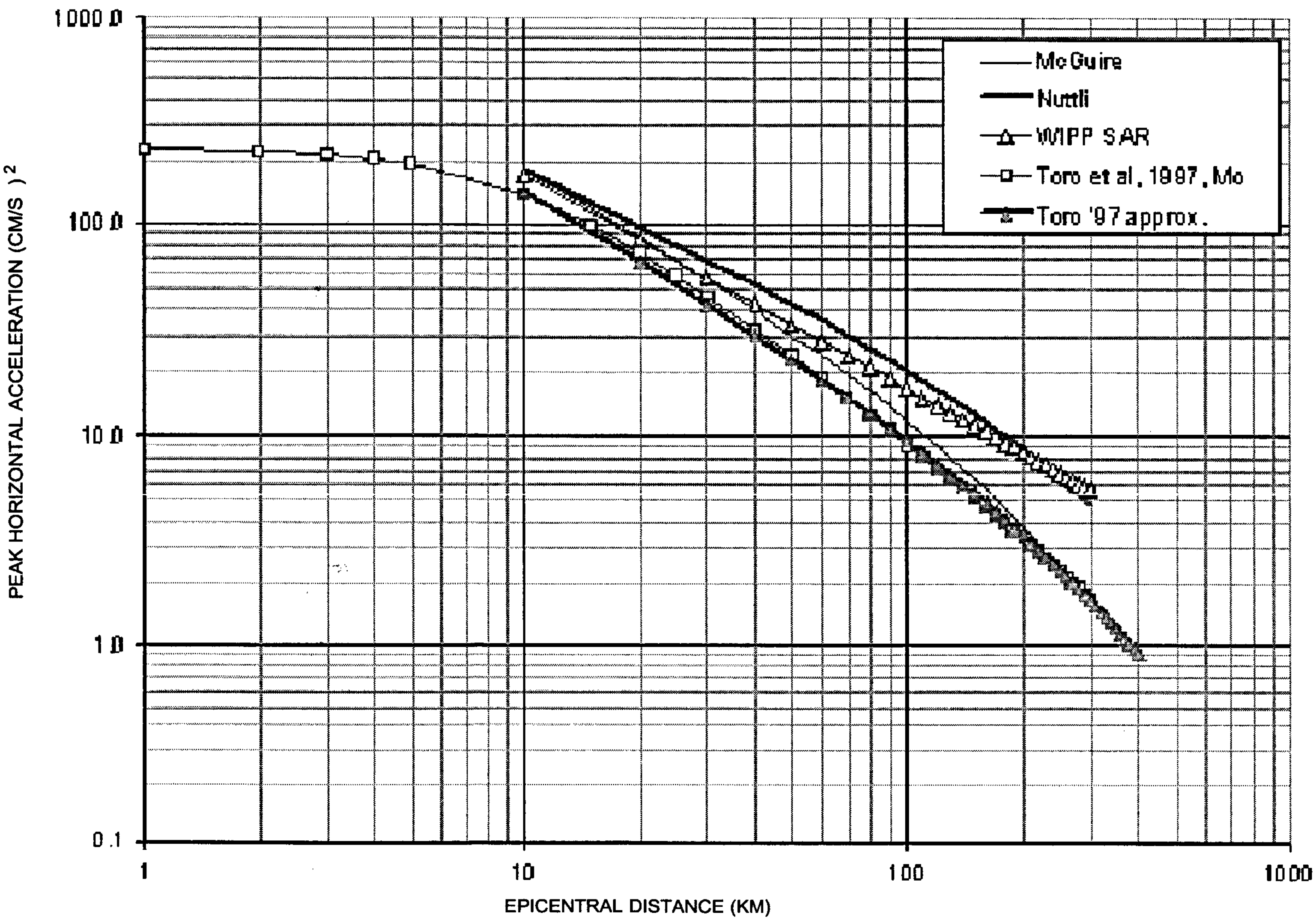
50 0 50 100 150 200 MILES

REFERENCE NUMBER
2Figures3.2.dwg

FIGURE 3.2-26

ALTERNATE SEISMIC SOURCE GEOMETRIES USED IN THE WIPP SEISMIC HAZARD STUDY

REVISION DATE: DECEMBER 2003

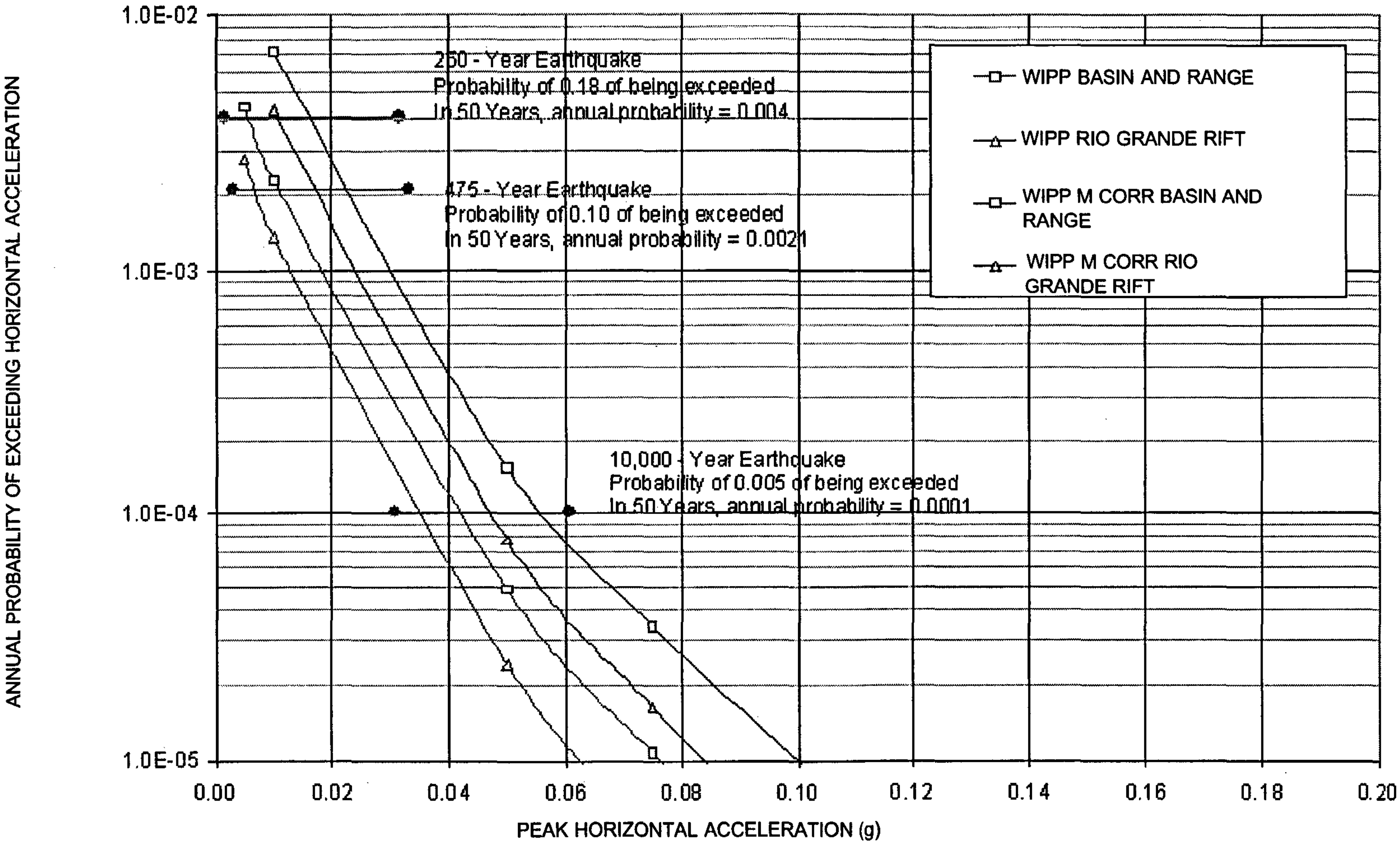


REFERENCE NUMBER
2Figures3.2.dwg

FIGURE 3.2-27

COMPARISION OF PGA ATTENUATION FOR A MAGNITUDE 5.0 EARTHQUAKE

REVISION DATE: DECEMBER 2003

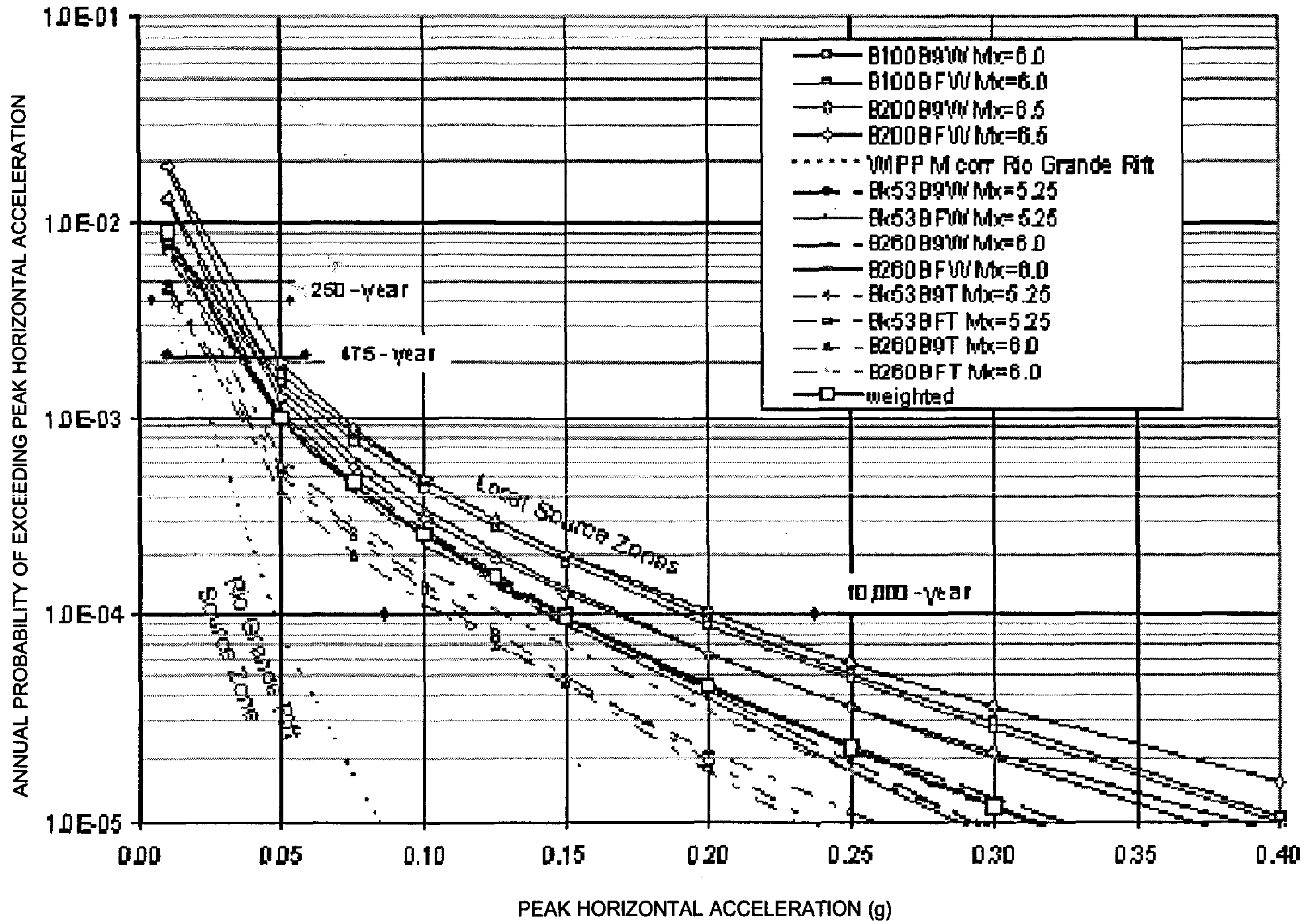


REFERENCE NUMBER
2Figures3.2.dwg

FIGURE 3.2-28

SEISMIC HAZARD AT THE NEF SITE FROM RIO GRANDE RIFT SEISMIC SOURCES

REVISION DATE: DECEMBER 2003

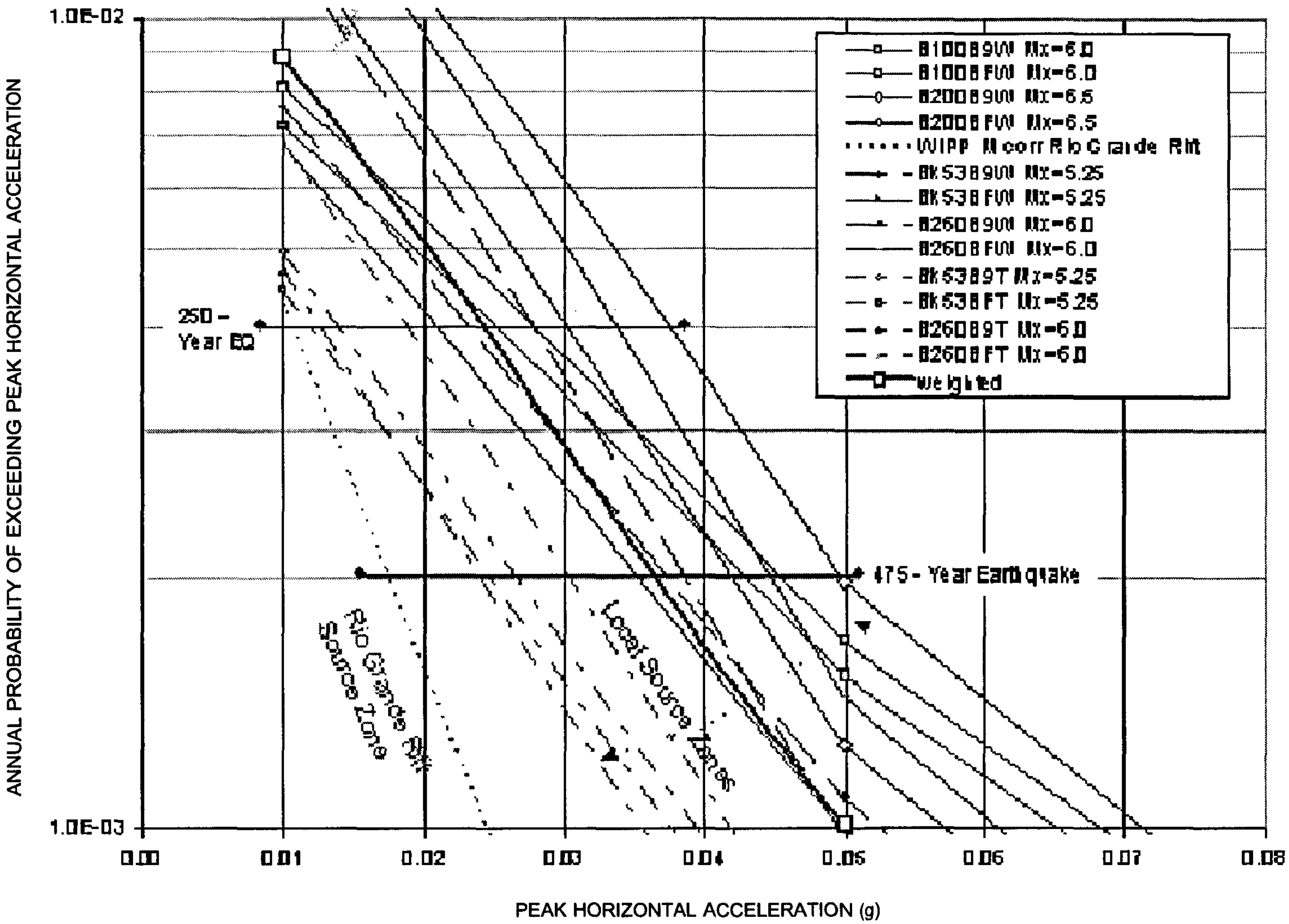


REFERENCE NUMBER
2Figures3.2.dwg

FIGURE 3.2-29

SEISMIC HAZARD AT THE NEF SITE FROM LOCAL SEISMIC SOURCE ZONES

REVISION DATE: DECEMBER 2003



0 100 200 300 KM

0 50 100 150 200 MILES

REFERENCE NUMBER
2Figures3.2.dwg

FIGURE 3.2-30

ZOOM OF SEISMIC HAZARD AT THE NEF SITE FROM LOCAL SEISMIC SOURCE ZONES

REVISION DATE: DECEMBER 2003

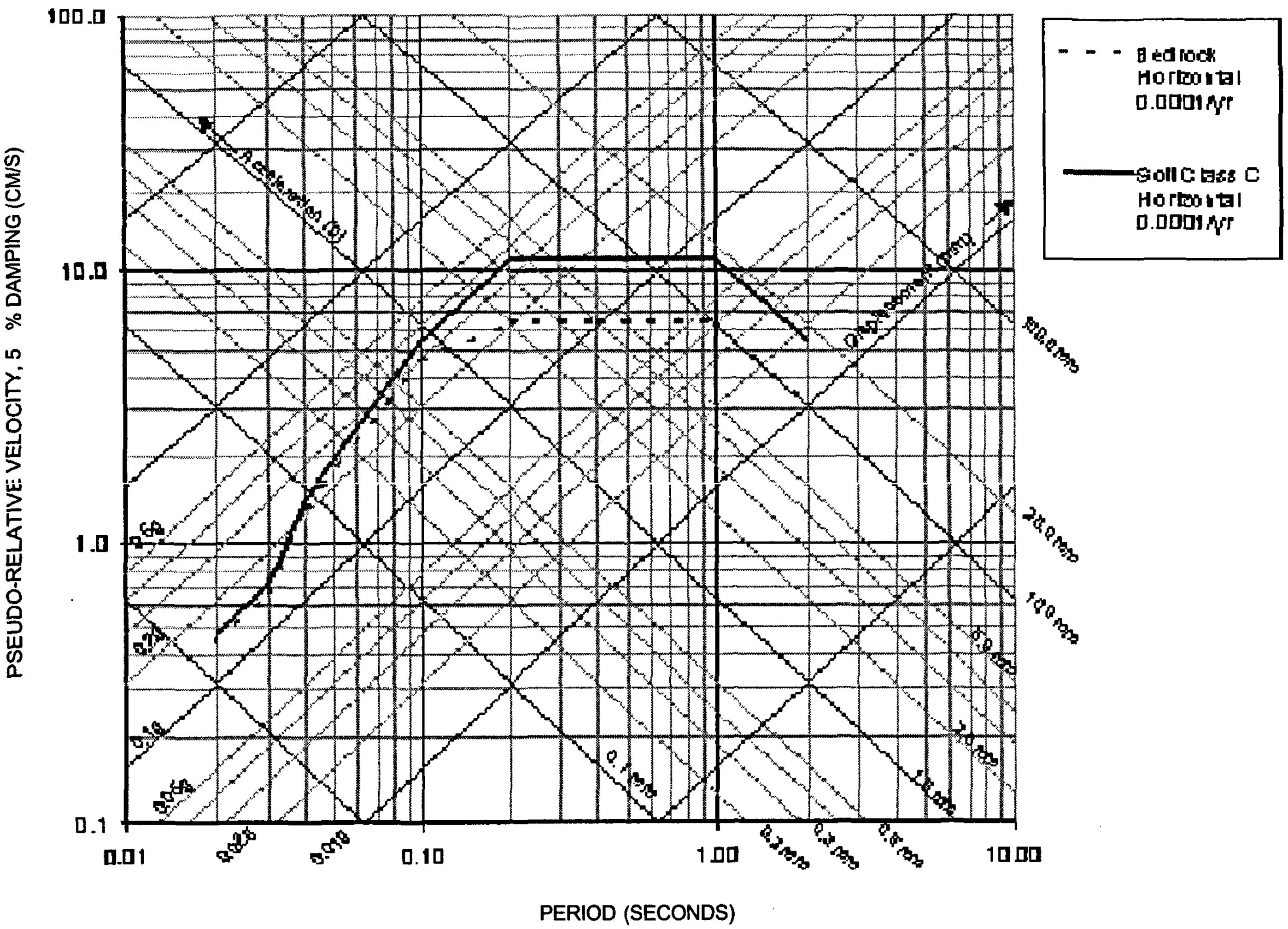
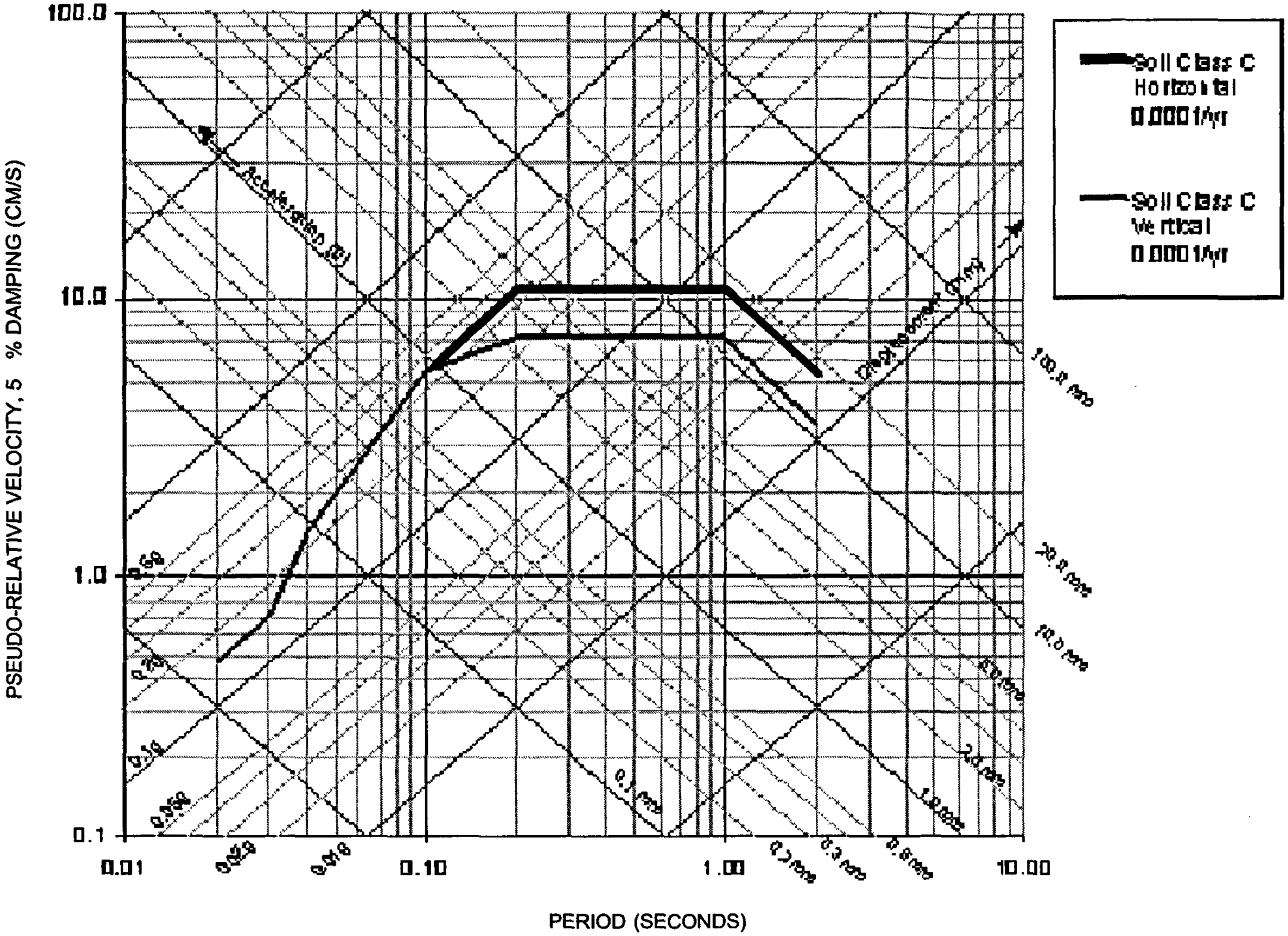


FIGURE 3.2-31

HORIZONTAL RESPONSE SPECTRA FOR THE 10,000-YEAR EARTHQUAKE – BEDROCK AND SOIL CLASS C FOR THE NEF SITE

REVISION DATE: DECEMBER 2003

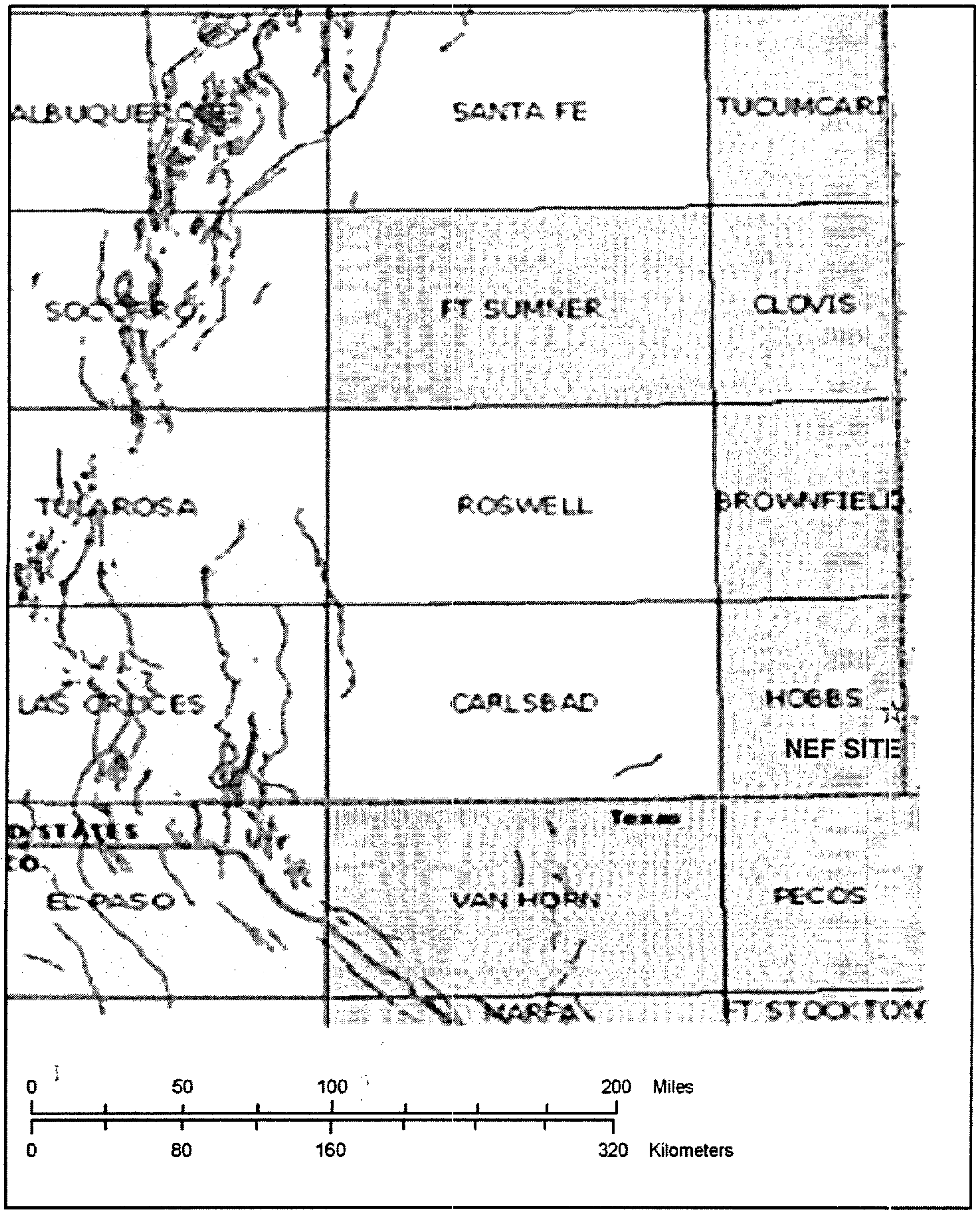
REFERENCE NUMBER
2Figures3.2.dwg



REFERENCE NUMBER
2Figures3.2.dwg

FIGURE 3.2-32
HORIZONTAL AND VERTICAL RESPONSE SPECTRA FOR THE 10,000-YEAR EARTHQUAKE SOIL CLASS C FOR THE NEF SITE

REVISION DATE: DECEMBER 2003

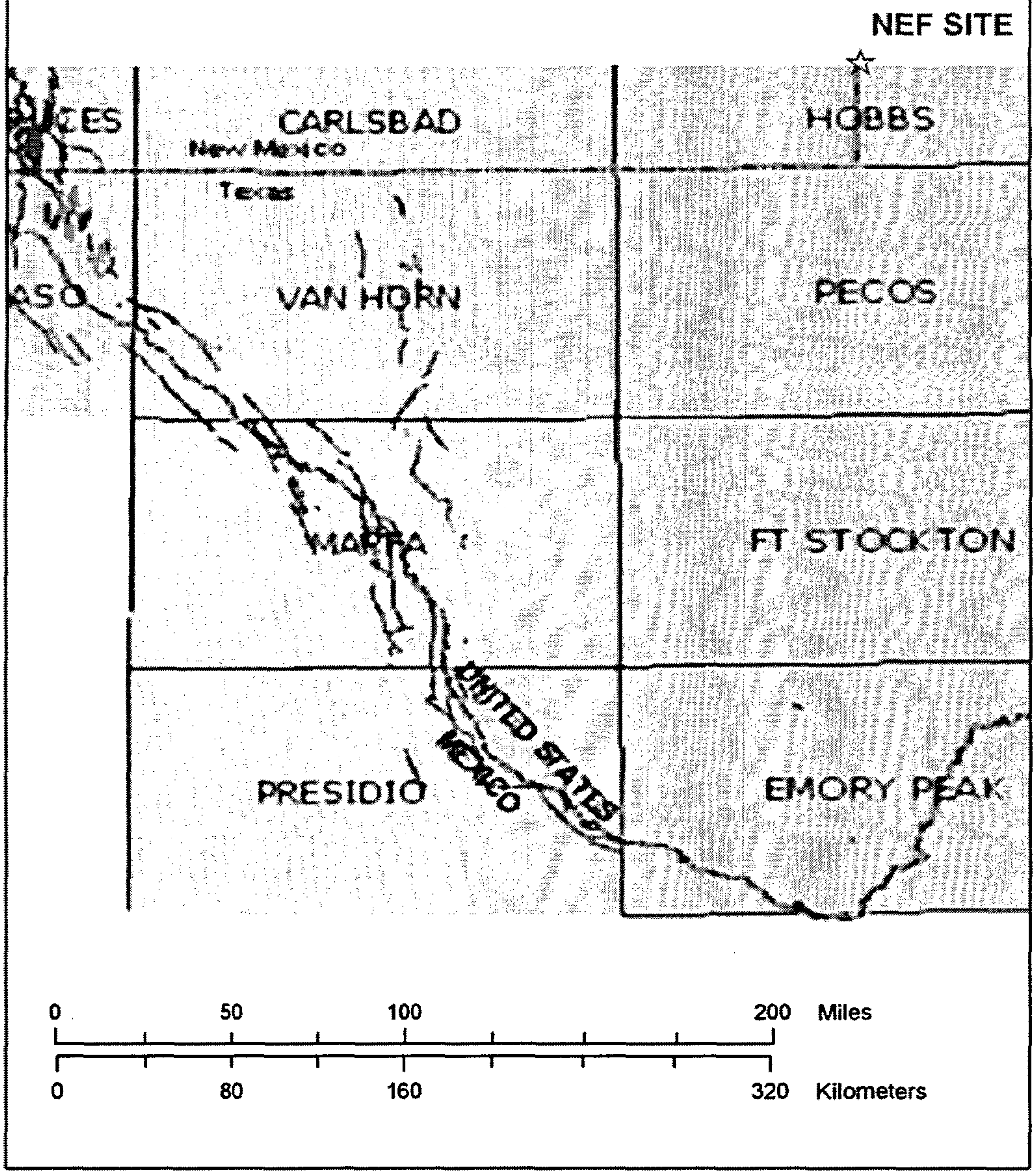


2

SOURCE: Earthquake Hazards Program
Quaternary Fault and Fold Database
(USGS, 2004)

FIGURE 3.2-33
QUATERNARY FAULTS
IN NEW MEXICO

REVISION 2 DATE: JULY 2004

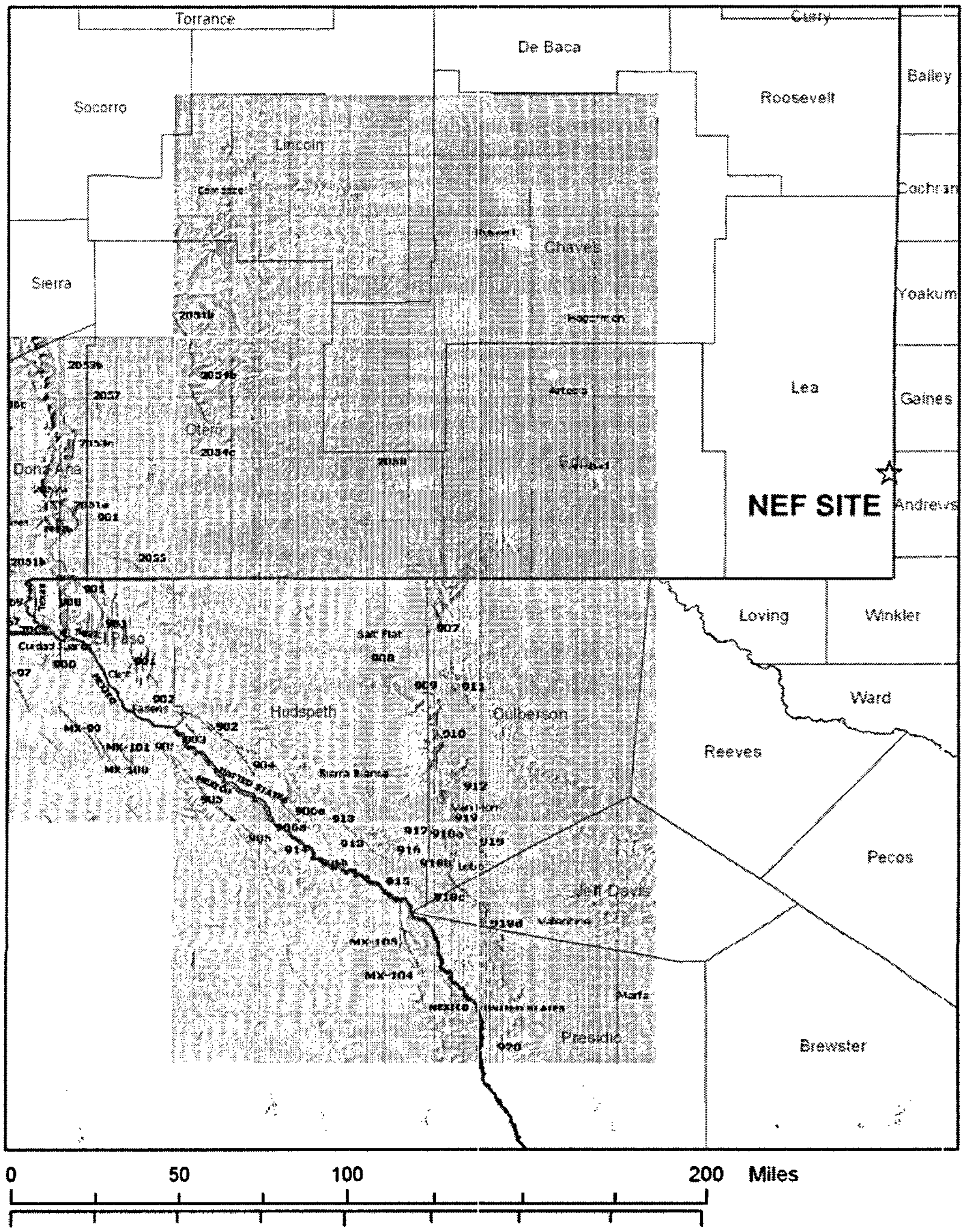


2

SOURCE: Earthquake Hazards Program
Quaternary Fault and Fold Database
(USGS, 2004)

FIGURE 3.2-34
QUATERNARY FAULTS
IN TEXAS

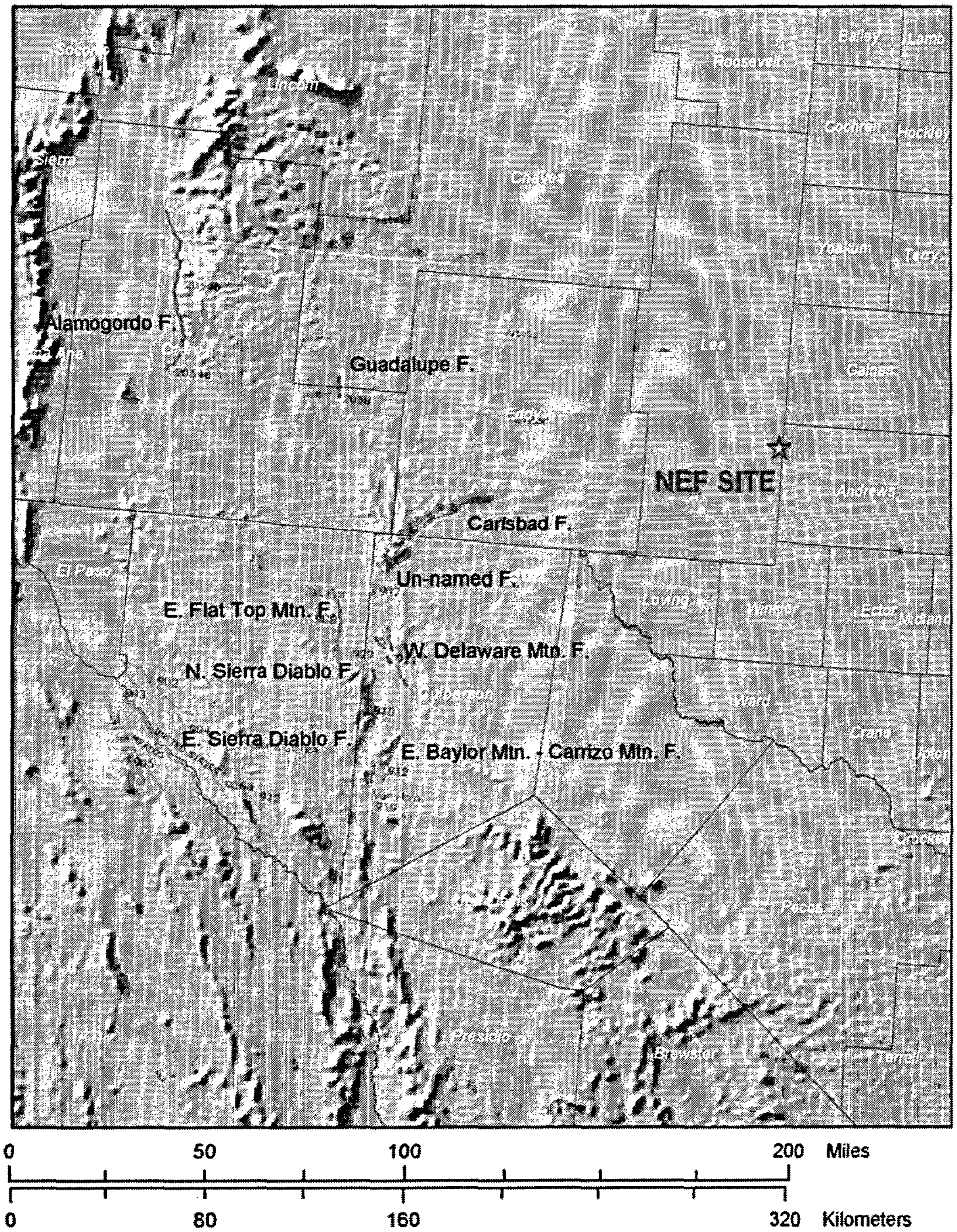
REVISION 2 DATE: JULY 2004



SOURCE: USGS, 2004; Machette, 2000

FIGURE 3.2-35
QUATERNARY FAULTS
WITHIN 322 km (200 mi) OF NEF SITE

REVISION 2 DATE: JULY 2004



NOTE: Locations of nearest capable faults (red traces) and older faults (blue traces)

FIGURE 3.2-36

LOCATIONS OF NEAREST FAULTS TO THE NEF SITE

REVISION 2 DATE: JULY 2004

2

NATIONAL
ENRICHMENT
FACILITY

INTEGRATED SAFETY ANALYSIS SUMMARY

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3.4

3

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3.3 FACILITY DESCRIPTION

The arrangement of the National Enrichment Facility (NEF) is shown in Figure 3.3-1, Facility Buildings and Areas. The major structures and functional areas of the facility are discussed in the following sections.

Distances from the facility to the site boundary were determined using guidance from U.S. NRC Regulatory Guide 1.145 (NRC, 1982), i.e., the nearest point on the building complex to the site boundary within a 45-degree sector centered on the compass direction of interest. These distances are provided in Table 3.3-1, Distances to Site Boundary and to Restricted Area Boundary and Wind Frequencies.

The distance to the nearest resident is greater than 4.26 km (2.63 mi).

3.3.1 Buildings and Major Components

3.3.1.1 Separations Building Modules

The overall layout of a Separations Building Module is presented in Figures 3.3-2 through 3.3-6. The facility includes three identical Separations Building Modules. Each module consists of two Cascade Halls, each of which houses a number of cascades connected in parallel producing a single product concentration at any one time. Each Cascade Hall is capable of producing 500,000 separative work units (SWU) per year. In addition to the Cascade Halls, each Separations Building Module houses a UF_6 Handling Area and a Process Services Area.

3.3.1.1.1 Design Description

Each Separations Building Module is approximately 170.0 m (557.75 ft) long x 67.9 m (222.75 ft) wide and 13.0 m (42.7 ft) high and totals 12,730 m^2 (137,025 ft^2), including both elevated floors of the Process Services Area. It is classified as a Special Purpose Industrial Occupancy area by the NFPA 101 (NFPA, 1997). It is classified as a Type I Unsprinklered Construction area by the New Mexico Building Code (NMBC, 1997) and as Type I Construction by NFPA 220 (NFPA, 1999). The thermal enclosure surrounding each assay (centrifuge) shall be constructed of and insulated with non-combustible materials (and is considered a fire barrier addressed by IROFS35).

Several chemical traps on the second floor of the Process Services Area contain hazardous materials. The chemical traps are housed in fire rated enclosures to meet the requirements of Section 6.4 of NFPA 101 (NFPA, 1997). The Separations Building Modules are designed to meet the occupant and exiting requirements set by NFPA 101 (NFPA, 1997) and to meet the construction type classifications set by the New Mexico Building Code (NMBC, 1997). The construction type and occupancy classification allow the Separations Building Modules to be unsprinklered. The UF_6 Handling Areas are separated from the Cascade Halls by one-hour fire-rated construction. The Separations Building Modules are also separated from each other by one-hour fire-rated construction.

3.3.1.1.2 Functional Areas and Major Components

3.3.1.1.2.1 Cascade Halls

Each Cascade Hall contains eight cascades. The centrifuges are mounted on precast concrete floor mounting elements (flomels). Each Cascade Hall is enclosed by a structural steel frame, which is supporting insulated sandwich panels. This cascade enclosure surrounds each Cascade Hall to aid in maintaining a constant temperature within the cascade enclosure. This enclosure will also be constructed to have a minimum one hour fire-resistive rating.

3.3.1.1.2.2 Process Services Area

The Process Services Area contains the gas transport equipment, which connects the cascades to the UF_6 Feed System, the Product Take-off System, the Tails Take-off System and the Contingency Dump System.

The first floor of the Process Services Area, at elevation 1,040 m (3,415 ft) mean sea level (msl), contains various pieces of equipment, control cabinets and electrical cabinets. The second floor of the Process Services Area, at elevation 1,045 m (3,431.5 ft) msl, contains various pieces of equipment, control cabinets, electrical cabinets, valve support frames, process pumps and chemical traps. The third floor of the Process Services Area at elevation 1,049 m (3,444.5 ft) msl, contains various pieces of equipment, control cabinets, electrical cabinets, water pumps and heating and ventilation equipment. The various floors of the Process Services Area can be accessed by one of three stairways or by the elevator.

A. UF_6 Handling Area

The UF_6 Handling Area contains the UF_6 Feed System, the Product Take-off System, and the Tails Take-off System. The UF_6 Handling Area is approximately 43.3 m (142 ft) x 67.9 m (222.75 ft) and totals 2940 m^2 (31,646 ft^2).

Rail transporters travel on rails embedded in the floor along the entire length of the facility. The rail transporter moves the cylinders to and from the appropriate feed or receiver stations. It has the ability to handle both the 48-inch feed cylinders and UBCs and 30-inch or 48-inch product cylinders.

3.3.1.1.2.3 Building Construction

Each Separations Building Module superstructure is structurally independent from the rest of the facility and is designed to be missile resistant. The superstructure is of precast/prestressed concrete construction using rectangular columns, rectangular and inverted tee beams, double or single tee roof and floor members and solid wall panels.

The roof structure over the Separations Building Module consists of deep precast/prestressed concrete double or single tee members covered with a thin layer of isocyanurate insulation board, which provides a barrier between the concrete surface and the single-ply roof membrane. The single ply membrane is then covered by 100 mm (4 in) of dow board insulation, filter fabric and concrete pavers. The tee members are supported by concrete 'L' girders around the perimeter and inverted tee girders on interior spans. These will, in turn, be supported by concrete columns supported on concrete spread footings. The roof assembly has a minimum combined thermal resistance value of R-20.

Exterior walls are precast insulated concrete panels. These walls will act as shear walls to provide lateral support for the structure. The exterior wall assembly has a minimum combined thermal resistance value of R-10. The interior side of the exterior wall is smooth concrete, which has been sealed and painted.

Interior non-load bearing walls are constructed of 200 mm (8 in) concrete block with an epoxy painted finish. These walls extend to the underside of the structure where required.

The floors of the Cascade Halls have a floor profile quality classification of flat in accordance with ACI 117-90 (ACI, 1990a) to aid in the transport of assembled centrifuges.

Floors in the Cascade Halls and UF_6 Handling Areas are of exposed concrete with a washable epoxy coating finish. The coatings are designed to resist process chemicals, decontamination agents and radiation.

3.3.1.2 Technical Services Building

The overall layout of the Technical Services Building (TSB) is presented in Figures 3.3-7 through 3.3-9. The TSB is located between column lines 1 and 11 and column lines N.1 and W, adjacent to the Blending and Liquid Sampling Area. The TSB contains support areas for the facility. It also acts as the secure point of entry to the Separations Building Modules and the Cylinder Receipt and Dispatch Building (CRDB).

3.3.1.2.1 Design Description

The TSB is a two-story structure, 10.0 m (32.8 ft) in height and totals 9,192 m^2 (98,942 ft^2). It is classified as a Special Purpose Industrial Occupancy area by NFPA 101 (NFPA, 1997). The TSB is classified as a Type I Unsprinklered Construction area by the New Mexico Building Code (NMBC, 1997) and as Type I Construction by NFPA 220 (NFPA, 1999). The TSB is designed to meet the occupant and exiting requirements set by the NFPA 101 (NFPA, 1997) and to meet the construction type classifications set by the New Mexico Building Code (NMBC, 1997). Several areas of the TSB have hazardous materials in quantities less than exempt amounts and are separated from areas by one-hour fire-rated construction. These areas include:

- Solid Waste Collection Room
- Vacuum Pump Rebuild Workshop
- Decontamination Workshop
- Ventilated Room.

Several of the TSB areas are separated from adjacent areas by one-hour fire-rated construction. These areas include:

- Liquid Effluent Collection and Treatment Room
- TSB GEVS Room
- Sample Storage Room which is located in the Chemical Lab.

3.3.1.2.2 Functional Areas and Major Components

3.3.1.2.2.1 Solid Waste Collection Room

The Solid Waste Collection Room is designed to process both wet and dry low-level radioactive solid waste. The Solid Waste Collection System is described in Section 3.5.13, Solid Waste Collection. Wet waste is categorized as radioactive, hazardous or industrial waste and includes assorted materials, oil recovery sludge, oil filters and miscellaneous hazardous wastes. Dry waste is also categorized as radioactive, hazardous or industrial waste and includes assorted materials, activated carbon, activated aluminum oxide, activated sodium fluoride, high efficiency particulate air (HEPA) filters, scrap metal and miscellaneous hazardous materials.

This room is approximately 15.0 m (49.25 ft) x 20.0 m (65.6 ft) x 5.0 m (16.4 ft) high and totals 300 m^2 (3,229 ft^2). It is classified as a Special Purpose Industrial Occupancy area with a less than exempt amount of hazardous materials. This area is separated from the other Special Purpose Industrial Occupancy areas by one-hour fire-rated construction.

3.3.1.2.2.2 Vacuum Pump Rebuild Workshop

The Vacuum Pump Rebuild Workshop is designed to provide space for the maintenance and re-building of plant equipment, mainly pumps which have been decontaminated in the Decontamination Workshop, and other miscellaneous plant equipment.

This room is approximately 12.8 m (42 ft) x 20.0 m (65.6 ft) x 5.0 m (16.4 ft) high and contains 256 m^2 (2,756 ft^2). The workshop consists of an open area, a storage area and a data logging/progress chasing area. It is equipped with suitable area lighting, a degassing oven, heating, ventilating, and air conditioning (HVAC), local extract systems, vacuum systems and a spray booth with a filter and extraction system. It is classified as a Special Purpose Industrial Occupancy area with a less than exempt amount of hazardous materials. This area is separated from the other Special Purpose Industrial Occupancy areas by one-hour fire-rated construction.

3.3.1.2.2.3 Decontamination Workshop

The purpose of the Decontamination Workshop is to provide a maintenance facility for both UF_6 pumps and vacuum pumps. It is also used for the temporary storage and subsequent dismantling of failed pumps. The activities carried out within the Decontamination Workshop include receipt and storage of contaminated pumps, out-gassing, Fomblin oil removal and storage, pump stripping, and the dismantling and maintenance of valves and other plant components.

The Decontamination Workshop also provides a facility for the removal of radioactive contamination from contaminated materials and equipment. The Decontamination System consists of a series of steps including equipment disassembly, degreasing, decontamination, drying and inspection. Components commonly decontaminated include pumps, valves, piping, instruments, sample bottles, tools and scrap metal. The Decontamination System is described in Section 3.5.14, Decontamination Workshop.

The Decontamination Workshop is maintained at a lower pressure than surrounding areas. Therefore any equipment or personnel entering this room must go through an air-lock.

This room is approximately 22.1 m (72.5 ft") x 20.0 m (65.6 ft) x 5.0 m (16.4 ft) high and contains 442 m^2 (4,758 ft^2). It is classified as a Special Purpose Industrial Occupancy area with a less than exempt amount of hazardous materials. This area is separated from the other Special Purpose Industrial Occupancy areas by one-hour fire-rated construction.

3.3.1.2.2.4 Ventilated Room

The Ventilated Room is designed to provide space for the maintenance of chemical traps and cylinders. The Ventilated Room is also used for the temporary storage of full and empty chemical traps and the contaminated chemicals used in the chemical traps.

The activities carried out within the Ventilated Room include receipt and storage of saturated chemical traps, chemical removal and temporary storage, contaminated cylinder pressure testing, and UF_6 cylinder pump out and valve maintenance.

The Ventilated Room is maintained at a lower pressure than surrounding areas. Therefore, any equipment or personnel entering this room must go through an air-lock.

This room is approximately 14.9 m (48.9 ft) x 20.0 m (65.6 ft) x 5.0 m (16.4 ft) high and contains 298 m^2 (3,208 ft^2). It is classified as a Special Purpose Industrial Occupancy area with a less than exempt amount of hazardous materials. This area is separated from the other Special Purpose Industrial Occupancy areas by one-hour fire-rated construction.

3.3.1.2.2.5 Cylinder Preparation Room

The Cylinder Preparation Room is designed for the purpose of testing and inspecting new or cleaned 30B, 48X, and 48Y cylinders for use in the facility.

This room is approximately 25.0 m (82 ft) x 20.0 m (65.6 ft) x 10 m (32.8 ft) high and totals 500 m^2 (5,382 ft^2). It is classified as a Special Purpose Industrial Occupancy area.

The Cylinder Preparation Room is maintained at a lower pressure than surrounding areas. Therefore any equipment or personnel entering this room must go through an air-lock.

3.3.1.2.2.6 Mechanical, Electrical and Instrumentation (ME&I) Workshop

The ME&I Workshop is designed to provide space for the normal maintenance of non-contaminated plant equipment. The facility also deals with faults associated with the pump motors, all instrument and control equipment, lighting, power, and associated process and services pipe work. It also provides space for the temporary storage of rebuilt equipment and other minor plant equipment.

This room is approximately 14.8 m (48.6 ft) x 20.0 m (65.6 ft) x 10.0 m (32.8 ft) high and totals 296 m^2 (3,186 ft^2). It is classified as a Special Purpose Industrial Occupancy area.

3.3.1.2.2.7 Liquid Effluent Collection and Treatment Room

The Liquid Effluent Collection and Treatment Room is designed for the collection of potentially contaminated liquid effluents produced on site, which are monitored for contamination prior to processing. These liquid effluents are stored in tanks prior to processing. The effluents are segregated into significantly contaminated effluent, slightly contaminated effluent or non-contaminated effluent. Liquid effluents produced by the facility include hydrolysed uranium hexafluoride and aqueous laboratory effluent, degreaser water, citric acid, laundry effluent water, floor washings, miscellaneous condensates and active area hand washings/shower water. The Liquid Waste Collection System is described in Section 3.5.12, Liquid Effluent Collection and Treatment System.

This room is approximately 19.8 m (64.9 ft) x 20.0 m (65.6 ft) x 10.0 m (32.8 ft) high and totals 396 m^2 (4,263 ft^2). It is classified as a Special Purpose Industrial Occupancy area. The Liquid Effluent Collection and Treatment Room is separated from adjacent areas by one-hour fire-rated construction.

3.3.1.2.2.8 Laundry

The Laundry is designed to clean contaminated and soiled clothing and other articles, which have been used throughout the facility. Laundry is sorted into two categories, articles with a high possibility of contamination and articles unlikely to have been contaminated. Those that are likely to be contaminated are further sorted into lightly and heavily soiled articles. Heavily soiled articles are transferred to the solid waste disposal system without having been washed.

The Laundry contains two industrial quality washing machines (75 kg (165 lb) capacity), two industrial quality dryers (75 kg (165 lb) capacity), one sorting hood to draw potentially contaminated air away, a sorting table and an inspection table. The Laundry System is described in Section 3.5.16, Laundry System. The Laundry also contains a small office and storage room.

This room is approximately 161.2 m^2 (1,735 ft^2). It is classified as a Special Purpose Industrial Occupancy area.

3.3.1.2.2.9 TSB Gaseous Effluent Vent System (GEVS) Room

The TSB GEVS is designed to remove UF_6, particulates containing uranium, and hydrogen fluoride (HF) from potentially contaminated process gas streams. Prefilters and High Efficiency Particulate Air filters remove particulates, including uranium particles, and impregnated and activated charcoal filters remove any residual traces of uranium and HF. The TSB GEVS is described in Section 3.4.9, Gaseous Effluent Vent System. The major components of the TSB GEVS are located in the TSB GEVS Room.

This room is approximately 9.6 m (31.5 ft) x 20.0 m (65.6 ft) x 10.0 m (32.8 ft) high and totals 192 m^2 (2,067 ft^2). It is classified as a Special Purpose Industrial Control area and is separated from the other Special Purpose Industrial Occupancy areas by one-hour fire-rated construction.

3.3.1.2.2.10 Mass Spectrometry Laboratory

The Mass Spectrometry Laboratory is designed for the purpose of measuring the isotopic abundance of various uranium isotopes in prepared samples, the bulk comprising hydrolysed uranium hexafluoride.

This room is approximately 10.3 m (33.75 ft) x 20.0 m (65.6 ft) and totals 206 m^2 (2,217 ft^2). It is classified as a Special Purpose Industrial Occupancy area.

3.3.1.2.2.11 Chemical Laboratory

The Chemical Laboratory is designed for the purpose of analyzing solid and liquid samples taken from all areas of the facility. It includes space for an analytical area, sub sampling area, wash area and weighing area.

This room is approximately 16.2 m (53.2 ft) x 20.0 m (65.6 ft) and totals 324 m^2 (3,488 ft^2). It is classified as a Special Purpose Industrial Occupancy area. The Sample Storage Room in the Chemical Laboratory is one-hour fire-rated construction.

3.3.1.2.2.12 Environmental Monitoring Laboratory

The Environmental Monitoring Laboratory is designed for the purpose of preparing and analyzing samples associated with safety or regulatory compliance.

This room and associated office space are approximately 17.3 m (56.75 ft) x 19.3 m (63.3 ft) and totals 334 m^2 (3,595 ft^2). It is classified as a Special Purpose Industrial Occupancy area.

3.3.1.2.2.13 Truck Bay/Shipping and Receiving Area

The Truck Bay is used as a place to load packaged low-level radioactive wastes onto trucks for transportation off site to a licensed processing facility or licensed disposal facility. It is also used for miscellaneous shipping and receiving.

This room is approximately 4.6 m (15.08 ft) x 9.8 m (32.2 ft) and totals 45 m^2 (484 ft^2). It is classified as a Special Purpose Industrial Occupancy area.

3.3.1.2.2.14 Medical Room

The Medical Room is designed to provide space for a nurse's station. This room is approximately 5.2 m (17 ft) x 5.4 m (17.75 ft) and totals 28 m^2 (301 ft^2). It is classified as a Special Purpose Industrial Occupancy area.

3.3.1.2.2.15 Radiation Monitoring Control Room

The Radiation Monitoring Control Room is designed to be the point of demarcation between non-contaminated areas and potentially contaminated areas of the facility. It includes space for a hand and foot monitor, hand washing facilities, safety showers, and boot barrier access.

This room is approximately 3.65 m (12 ft) x 8.4 m (27.6 ft) and totals 30 m^2 (323 ft^2). It is classified as a Special Purpose Industrial Occupancy area.

3.3.1.2.2.16 Break Room

The Break Room has room for vending machines, tables and a small kitchenette. It also serves as an assembly area for emergency planning purposes and has area allocated for the storage of emergency equipment and supplies and emergency monitoring equipment.

This room is approximately 7.3 m (23.9 ft) x 15.0 m (49.25 ft) and totals 110 m^2 (1,184 ft^2). It is classified as a Special Purpose Industrial Occupancy area.

3.3.1.2.2.17 Control Room

The Control Room and associated support area are approximately 14.4 m (47.25 ft) x 12.6 m (41.3 ft) and totals 181 m^2 (1,948 ft^2) and is the main monitoring point for the entire facility. It is classified as a Special Purpose Industrial Occupancy area. The Control Room provides all of the facilities for the control of the plant, operational requirements and personnel comfort. It is a permanently manned area and contains the following equipment:

- Overview screen
- Control desk
- Fire alarm system
- Storage facilities
- Communication systems.

The Plant Control Systems and the Communications and Alarms System are described in Section 3.5.9, Control Systems and Section 3.5.7, Communication and Alarm Annunciation Systems, respectively.

3.3.1.2.2.18 Training Room

The Training Room and associated support area are approximately 9.7 m (31.8 ft) x 10.6 m (34.75 ft) and totals 103 m^2 (1,108 ft^2) and is used for Control Room training. It is classified as a Special Purpose Industrial Occupancy area. It has visual and personnel access to the Control Room and contains the following:

- Plant Control System training system
- Centrifuge Monitoring System training system
- Central Control System switches and servers.

3.3.1.2.2.19 Security Alarm Center

The Security Alarm Center is approximately 7.0 m (23 ft) x 5.6 m (18.3 ft) and totals 39 m^2 (420 ft^2) and is used as the primary security monitoring station for the facility. It is classified as a Special Purpose Industrial Occupancy area. All electronic security systems are controlled and monitored from this center. These systems include Closed Circuit Television (CCTV), Intrusion Detection and Assessment (IDA), Access Control and radio dispatch.

3.3.1.2.3 Building Construction

The TSB superstructure is of precast/prestressed concrete construction using rectangular columns, rectangular and inverted tee beams, double or single tee roof and floor members and solid wall panels.

The roof structure over the TSB consists of deep precast/prestressed concrete double or single tee members covered with a thin layer of isocyanurate insulation board that provides a barrier between the concrete surface and the single-ply roof membrane. The single ply membrane is then covered by 100 mm (4 in) of dow board insulation, filter fabric and concrete pavers. The tee members are supported by concrete 'L' girders around the perimeter and inverted tee girders on interior spans. These, in turn, are supported by concrete columns supported on concrete spread footings. The roof assembly has a minimum combined thermal resistance value of R-20.

Exterior walls are precast insulated concrete panels. These walls act as shear walls to provide lateral support for the structure. The exterior wall assembly has a minimum combined thermal resistance value of R-10. The interior side of the exterior wall is of smooth concrete that has been sealed and painted. Interior non-load bearing walls are constructed of 200 mm (8 in) concrete block with an epoxy painted finish. These walls extend to the underside of the structure where required.

Floors in the TSB technical areas are of exposed concrete with a washable epoxy coating finish. The coatings are designed to resist process chemicals, decontamination agents and radiation.

3.3.1.3 Cylinder Receipt and Dispatch Building (CRDB)

The overall layout of the CRDB is presented in Figures 3.3-10 through 3.3-12. The CRDB is located between two Separations Building Modules, adjacent to the Blending and Liquid Sampling Area.

3.3.1.3.1 Design Description

The CRDB is approximately 45.9 m (150.6 ft) wide x 246.2 m (807.75 ft) long and 13.0 m (42.7 ft) high and totals 11,300 m^2 (121,638 ft^2). The entire CRDB is open to the underside of the roof. It is classified as a Storage Occupancy area by the NFPA 101 (NFPA, 1997). It is classified as a Type I Unsprinklered Construction area by the New Mexico Building Code (NMBC, 1997) and as Type I Construction by NFPA 220 (NFPA, 1999). The CRDB is designed to meet the occupant and exiting requirements set by the NFPA 101 (NFPA, 1997) and to meet the construction type classification set by the New Mexico Building Code (NMBC, 1997). The CRDB is separated from the separations modules and Blending and Liquid Sampling Area by one-hour fire-rated construction. The CRDB exterior walls are a minimum one-hour fire-rated construction.

3.3.1.3.2 Functional Areas and Major Components

All UF_6 feed cylinders and empty product cylinders and uranium byproduct cylinders (UBCs) enter the facility through the CRDB. It is designed to include space for the following:

- Loading and unloading of cylinders
- Inventory weighing
- Buffer storage of feed cylinders
- Preparation and storage of overpack protective packaging
- Semi-finished product storage
- Final product storage
- Prepared cylinder storage.

The majority of the floor area is used as lay-down space for the cylinders, for both storage and preparation. The cylinders are placed on specially designed cradles called stillages to stabilize them while being stored in the CRDB.

Cylinders are delivered to the facility in transport trucks. The trucks enter the CRDB through the main vehicle loading bay, located between column lines 40 and 41, which is equipped with vehicle access platforms that aid with cylinder loading and unloading. Two double girder bridge cranes handle the cylinders within the CRDB. Each crane spans 1/2 the width and runs the full length of the building.

After delivery, the cylinders are processed for receipt as either empty UBCs (48-in cylinders) or empty product cylinders (30-in or 48-in cylinders) or UF_6 feed cylinders (48-in cylinders). They are inspected and weighed and moved to their appropriate locations. UF_6 feed cylinders are delivered to a storage area in the CRDB.

When required for processing, the cylinders, which have been placed in storage areas are moved by the overhead cranes to the rail transporter located between column lines 15.4 and 16 of the CRDB. The CRDB rail transporter transports cylinders to the main rail transporter in the Blending and Liquid Sampling Area, which then delivers the cylinders to their required locations throughout the facility. Cylinders are removed from the facility in the same fashion.

3.3.1.3.3 Building Construction

The CRDB superstructure is designed to be missile resistant and is of precast/prestressed concrete construction using rectangular columns, rectangular and inverted tee beams, double or single tee roof and floor members and solid wall panels.

The two double girder bridge cranes are supported by a steel girder crane runway, supported by the precast concrete columns.

The roof structure over the CRDB consists of deep precast/prestressed concrete double or single tee members covered with a thin layer of isocyanurate insulation board that provides a barrier between the concrete surface and the single-ply roof membrane. The single ply membrane is then covered by 100 mm (4 in) of dow board insulation, filter fabric and concrete pavers. The tee members are supported by concrete 'L' girders around the perimeter and inverted tee girders on interior spans. These, in turn, are supported by concrete columns supported on concrete spread footings. The roof assembly has a minimum combined thermal resistance value of R-20.

Exterior walls are precast insulated concrete panels. These walls act as shear walls to provide lateral support for the structure. The exterior wall assembly has a minimum combined thermal resistance value of R-10. The interior side of the exterior wall is smooth concrete, which has been sealed and painted. Interior non-load bearing walls are constructed of 200 mm (8 in) concrete block with an epoxy painted finish. These walls extend to the underside of the structure where required.

The floor areas of the CRDB, which are used as a part of the centrifuge transport path, have a floor profile quality classification of flat in accordance with ACI 117-90 (ACI, 1990a) to aid in the transport of assembled centrifuges.

Floors in the CRDB are of exposed concrete with a washable epoxy coating finish. The coatings are designed to resist process chemicals, decontamination agents and radiation.

3.3.1.4 Centrifuge Assembly Building

The overall layout of the Centrifuge Assembly Building (CAB) is presented in Figures 3.3-13 through 3.3-16. The Centrifuge Assembly Building is located adjacent to the Cylinder Receipt and Dispatch Building.

3.3.1.4.1 Design Description

The CAB is approximately 50.9 m (167 ft) wide x 195.5 m (641.4 ft) long and ranges from 11 m (36.08 ft) to 16 m (52.5 ft) high. It totals approximately 11,364 m^2 (122,322 ft^2). The entire CAB is open to the underside of the roof. It is classified as a Special Purpose Industrial Occupancy area by NFPA 101 (NFPA, 1997). It is classified as a Type I Unsprinklered Construction area by the New Mexico Building code (NMBC, 1997). The CAB is designed to meet the occupant and exiting requirements set by NFPA 101 (NFPA, 1997) and to meet the construction type classifications set by the New Mexico Building code (NMBC, 1997) and as Type I Construction by NFPA 220 (NFPA, 1999). The CAB is separated from the CRDB one-hour fire-rated construction.

The Centrifuge Assembly Building is used for the assembly, inspection and mechanical testing of the centrifuges prior to installation in the Cascade Halls of the Separations Building Modules and introduction of UF_6. Centrifuge assembly operations are undertaken in clean room conditions. The building is divided into the following distinct areas:

- Centrifuge Component Storage Area
- Centrifuge Assembly Area 'A'
- Centrifuge Assembly Area 'B'
- Assembled Centrifuge Storage Area
- Building Office Area
- Centrifuge Test and Post Mortem Facilities.

3.3.1.4.2 Functional Areas and Major Components

3.3.1.4.2.1 Centrifuge Component Storage Area

The Centrifuge Component Storage Area serves as the initial receipt location for the centrifuge parts. It is designed to store up to four weeks stock of centrifuge components delivered from Europe. These components are delivered by truck in specifically designed containers, which are then packed into International Organization for Standardization (ISO) freight containers. The containers are off-loaded via fork lift truck and placed in the storage area through one of two roll up doors located at the end of the CAB.

Because the assembly operations are undertaken in clean room conditions, the centrifuge component containers are cleaned in a washing facility located within the Centrifuge Component Storage Area, prior to admission to the Centrifuge Assembly Area. The Centrifuge Component Storage Area also acts as an acclimatization area to allow components to equilibrate with the climatic conditions of the Centrifuge Assembly Area.

Transfer of components and personnel between the Centrifuge Component Storage Area and the Centrifuge Assembly Area is via an airlock to prevent ingress of airborne contaminants.

3.3.1.4.2.2 Centrifuge Assembly Area

Centrifuge components are assembled into complete centrifuges in this area. Assembly operations are carried out on two parallel production lines, A and B.

The centrifuge operates in a vacuum, therefore, centrifuge assembly activities are undertaken in clean room conditions, ISO Class 5 according to ISO 14644-1:1999E (ISO, 1999), to prevent ingress of volatile contaminants which would have a detrimental effect on centrifuge performance. Prior to installation into the cascade, the centrifuge has to be conditioned, which is done in the Centrifuge Assembly Area prior to storage in the Assembled Centrifuge Storage Area.

Local jib cranes are installed in certain areas and impose less than a 500 kg (1100 lb) load. The Centrifuge Assembly Area is separated from other areas by one-hour fire-rated construction.

3.3.1.4.2.3 Assembled Centrifuge Storage Area

Assembled and conditioned centrifuges are stored in the Assembled Centrifuge Storage Area prior to installation.

During construction of the facility, a separate installation team will access this area and transfer the assembled and conditioned centrifuges to the Cascade Halls for deployment.

Centrifuges are routed via a covered corridor that links the Assembled Centrifuge Storage Area with the CRDB. The covered corridor has the same standard of floor as the Assembled Centrifuge Storage Area.

3.3.1.4.2.4 Building Office Area

A general office area is located adjacent to the Centrifuge Assembly Area. It contains the main personnel entrance to the building as well as entrances to the Centrifuge Component Storage Area and Centrifuge Assembly Area. It is a two-story area that includes the following:

- Offices
- Change Rooms - The change rooms provide space where employees can dress in protective clothing as required
- Break Room
- Maintenance Area
- Chemical Storage Area
- Battery Charging Area.

3.3.1.4.2.5 Centrifuge Test and Post Mortem Facilities

The Centrifuge Test Facility is designed to:

- Provide a means of functionally testing the performance of production centrifuges to ensure compliance with design parameters
- Investigate production and operational problems.

This area consists of two test positions. The Centrifuge Post Mortem Facility is designed for investigating problems with production centrifuges. Based on 30 years of European experience, the demand for centrifuge post mortems is infrequent.

The principal functions of the Centrifuge Post Mortem Facility are:

- To facilitate dismantling of contaminated centrifuges using equipment and processes, which minimize the potential to contaminate personnel or adjacent facilities
- To prepare potentially contaminated components and materials for transfer to the TSB prior to disposal.

Centrifuges are brought into the facility on a specially designed transport cart via an airlock entry. The facility is also equipped with radiological monitoring devices, toilets and washing facilities, and hand, foot and clothing personnel monitors to detect surface contamination.

The Centrifuge Post Mortem Facility includes a centrifuge dismantling area and an inspection area. The centrifuge dismantling area includes a stand onto which the centrifuge to be dismantled is mounted providing access to the top and bottom of the centrifuge. A local jib crane is located over the stand to enable removal of the centrifuge from the transport cart and facilitate loading onto the stand. The inspection area includes an inspection bench, portable lighting, a microscope, an endoscope and a digital video/camera.

3.3.1.4.3 Building Construction

The CAB superstructure is designed of precast/prestressed concrete construction using rectangular columns, rectangular and inverted tee beams, double or single tee roof and floor members and solid wall panels.

The roof structure over the CAB consists of deep precast/prestressed concrete double or single tee members covered with a thin layer of isocyanurate insulation board that provides a barrier between the concrete surface and the single-ply roof membrane. The single ply membrane is then covered by 100 mm (4 in) of dow board insulation, filter fabric and concrete pavers. The tee members are supported by concrete 'L' girders around the perimeter and inverted tee girders on interior spans. These will, in turn, be supported by concrete columns supported on concrete spread footings. The roof assembly has a minimum combined thermal resistance value of R-20.

Exterior walls are precast insulated concrete panels. These walls act as shear walls to provide lateral support for the structure. The exterior wall assembly has a minimum combined thermal resistance value of R-10. The interior side of the exterior wall is smooth concrete that has been sealed and painted.

Interior non-load bearing walls are constructed of 200 mm (8 in) concrete block with an epoxy painted finish. These walls extend to the underside of the structure where required.

The floors of the CAB Assembled Centrifuge Storage Area have a floor profile quality classification of flat in accordance with ACI 117-90 (ACI, 1990a) to aid in the transport of assembled centrifuges.

Floors in the CAB are of exposed concrete with a washable epoxy coating finish. The coatings are designed to resist process chemicals, decontamination agents and radiation.

The Centrifuge Test Facility Area is separated from other areas by one-hour fire-rated construction.

3.3.1.5 Blending and Liquid Sampling Area

The Blending and Liquid Sampling Area is shown in Figure 3.3-17. The Blending and Liquid Sampling Area is adjacent to the CRDB and is located between two Separations Building Modules.

3.3.1.5.1 Design Description

The Blending and Liquid Sampling Area is approximately 45.9 m (150.6 ft) wide x 33.5 m (109.9 ft) long and 10.0 m (32.8 ft) high and totals 1,538 m^2 (16,555 ft^2). The entire area is open to the underside of the roof. It is classified as a Special Purpose Industrial Occupancy area by NFPA 101 (NFPA, 1997). It is classified as a Type I Unsprinklered Construction area by the New Mexico Building code (NMBC, 1997) and as Type I Construction by NFPA 220 (NFPA, 1999). The Blending and Liquid Sampling Area is designed to meet the occupant and exiting requirements set by the NFPA 101 (NFPA, 1997) and to meet the construction type classification set by the New Mexico Building code (NMBC, 1997). The Blending and Liquid Sampling Area is separated from the UF_6 Handling Areas by one-hour fire-rated construction.

3.3.1.5.2 Functional Areas and Major Components

The primary function of the Blending and Liquid Sampling Area is to provide means to fill 30B cylinders with UF_6 at a required ^{235}U concentration level and to obtain samples of the homogenized liquid UF_6. The area contains the major components associated with the Product Blending System and the Product Liquid Sampling System. The Product Blending System is described in Section 3.4.6, Product Blending System. The Product Liquid Sampling System is described in Section 3.4.7, Product Liquid Sampling System.

3.3.1.5.3 Building Construction

The Blending and Liquid Sampling Area superstructure is designed to be missile resistant and is of precast/prestressed concrete construction using rectangular columns, rectangular and inverted tee beams, double or single tee roof and floor members and solid wall panels.

The roof structure over the Blending and Liquid Sampling Area consists of deep precast/prestressed concrete double or single tee members covered with a thin layer of isocyanurate insulation board that provides a barrier between the concrete surface and the single-ply roof membrane. The single ply membrane is then covered by 100 mm (4 in) of dow board insulation, filter fabric and concrete pavers. The tee members are supported by concrete 'L' girders around the perimeter and inverted tee girders on interior spans. These, in turn, are supported by concrete columns supported on concrete spread footings. The roof assembly has a minimum combined thermal resistance value of R-20.

Exterior walls are precast insulated concrete panels. These walls act as shear walls to provide lateral support for the structure. The exterior wall assembly has a minimum combined thermal resistance value of R-10. The interior side of the exterior wall is smooth concrete, which has been sealed and painted.

Interior non-load bearing walls are constructed of 200 mm (8 in) concrete block with an epoxy painted finish. These walls extend to the underside of the structure where required.

Floors in the Blending and Liquid Sampling Area are of exposed concrete with a washable epoxy coating finish. The coatings are designed to resist process chemicals, decontamination agents and radiation.

3.3.1.6 Uranium Byproduct Cylinder (UBC) Storage Pad

The facility utilizes an area outside of the CRDB for storage of UBCs, which contain UF_6 that is depleted in ^{235}U. The tails are stored under vacuum in corrosion resistant Type 48Y cylinders. The UBC Storage Pad will also be used to store empty feed cylinders that are not immediately reconnected to the facility. The UBC Storage Pad is shown on Figure 3.3-1, Facility Buildings and Areas.

3.3.1.6.1 Design Description

The UBC Storage Pad is designed to provide storage for UBCs and six months of empty feed cylinders. Approximately 625 UBC per year are filled for storage. The UBC Storage pad is sized to accommodate 15,727 cylinders (capacity equivalent to 30 years of facility operation).

These cylinders are stacked two high. Concrete saddles are used to store the cylinders approximately 200 mm (8 in) above ground level. The UBC Storage Pad occupies approximately 8.50 ha (21 acres).

3.3.1.6.2 Functional Areas and Major Components

The UBC Storage Pad layout is based on moving the cylinders with cranes and flatbed trucks. Flatbed trucks are used to move the cylinders from the CRDB to the UBC Storage Pad. A double girder Gantry crane is used to remove the cylinders from the flatbed trucks and place them in the UBC Storage Pad. The Gantry crane is designed to double stack the cylinders in the storage area.

3.3.1.6.3 Construction

The UBC Storage Pad is constructed of a concrete pad with a dedicated collection and drainage system. Vehicle crash barriers are located along the site roads outside of the Controlled Access Area adjacent to the storage area. The entire area is fenced for security and radiological protection purposes.

3.3.1.7 Central Utilities Building

The Central Utilities Building (CUB) is shown on Figure 3.3-18.

3.3.1.7.1 Design Description

The CUB is approximately 24.8 m (81.3 ft) wide x 80.8 m (265.08 ft) long and 10 m (32.8 ft) high and totals 1962 m^2 (21,119 ft^2). It is classified as a Special Purpose Industrial Occupancy by NFPA 101 (NFPA, 1997). It is classified as a Type IIIN Unprotected, Sprinklered Construction area by the New Mexico Building Code (NMBC, 1997) and as Type II Construction by NFPA 220 (NFPA, 1999). The Central Utilities Building is designed to meet the occupant and exiting requirements set by the NFPA 101 (NFPA, 1997) and set by the New Mexico Building Code (NMBC, 1997).

3.3.1.7.2 Functional Areas and Major Components

The Central Utilities Building houses two diesel generators, which provide the site with standby power. The Standby Generator System is discussed in Section 3.5.10, Standby Diesel Generator System. The building contains day tanks, switchgear, and control panels. The rooms housing the diesels are constructed independent of each other with adequate provisions made for maintenance, equipment removal and equipment replacement, by roll-up and access doors.

The diesel fuel unloading area provides tanker truck access to the two above ground tanks, which provide diesel fuel storage. Secondary containment is provided to contain spills or leaks from the above ground diesel fuel tanks.

The CUB also houses the cooling water chillers and pumps, boiler room, hot water boilers and pumps, deionized water systems and air compressors. These systems are described in Sections 3.5.5, Cooling Water System, 3.5.4, Water Supply, and 3.5.3, Compressed Air System, respectively.

3.3.1.7.3 Building Construction

The Central Utilities Building superstructure is designed of structural steel framing.

The roof structure consists of metal decking over structural steel framing. The metal decking is covered with a built-up roof system. The roof assembly has a minimum combined thermal resistance value of R-20.

Exterior walls consist of a metal panel system. The exterior wall assembly has a minimum combined thermal resistance value of R-10.

Interior non-load bearing walls are constructed of 200 mm (8 in) concrete block with an epoxy painted finish. These walls extend to the underside of the structure where required.

Floors consist of exposed concrete with a washable epoxy coating finish.

3.3.1.8 Administration Building

3.3.1.8.1 Design Description

The Administration Building is near the TSB. It is approximately 1403 m^2 (15,102 ft^2) and 6.0 m (19.8 ft) high. It is classified as a New Business Occupancy area by the NFPA 101 (NFPA,1997) and is classified as a Type IIIN Unprotected Construction area by the New Mexico Building Code (NMBC, 1997) and as Type II Construction by NFPA 220 (NFPA, 1999). The Administration Building is designed to meet the occupant, and exiting requirements set by the NFPA 101 (NFPA,1997) and by the New Mexico Building Code (NMBC, 1997). The entire building is sprinklered.

3.3.1.8.2 Functional Areas and Major Components

The general office areas and the Entry Exit Control Point (EECP) for the facility are located in the Administration Building. All personnel access to the facility occurs at this location. Vehicular traffic passes through a security checkpoint before being allowed to park. Parking is located outside of the Controlled Access Area (CAA) security fence. Personnel enter the Administration Building and general office areas via the main lobby.

Personnel requiring access to facility areas or the CAA must pass through the EECP. The EECP is located at the rear of the main lobby and is designed to facilitate and control passage of authorized facility personnel and visitors to and from the CAA. Personnel entering the security Controlled Access Area are required to undergo, at a minimum, the following security screening at the EECP:

- Positive Identification – photo badge and/or biometrics
- Verification of access authorization

- Inspection of persons for unauthorized material (pass through a magnetometer)
- Inspection of all hand carried packages (x-ray screening).

In the main lobby, employees receive their badges and proceed through a turnstile into the office area or the EECP. Visitors check-in at the main lobby, where a receptionist notifies plant personnel of their arrival.

Entry to the facility areas from the Administration Building is only possible through the EECP.

Approximately 50 work locations are provided for the plant office staff. The office environment consists of private, semiprivate, and open office space. The lobby is designed to also act as an assembly area for emergency planning purposes. Area has been allocated for the storage of emergency equipment and supplies and emergency monitoring equipment. It also contains a kitchen, break room, conference rooms, and building service facilities such as a mechanical equipment room. An open office layout allows for flexibility in space allocation.

3.3.1.8.3 Building Construction

The Administration Building superstructure is designed of structural steel framing.

The roof structure consists of metal decking over structural steel framing. The metal decking is covered with a built-up roof system. The roof assembly has a minimum combined thermal resistance value of R-20.

Exterior walls consist of a combination of architectural metal panels and a curtain wall glazing system. The exterior wall assembly has a minimum combined thermal resistance value of R-10. The interior side of the exterior wall is faced with 16 mm (5/8 in) gypsum wallboard.

Interior non-load bearing walls are constructed of 92 mm (4 in) metal studs filled with batt insulation and faced with 16 mm (5/8 in) gypsum wallboard. Walls extend to 150 mm (6 in) above the ceiling or to the underside of the structure where required.

3.3.1.9 Visitor Center

A Visitor Center is located outside of the security fence area.

3.3.1.10 Site Security Buildings

3.3.1.10.1 Design Description

The main Security Building is located at the entrance to the facility. It functions as a security checkpoint for incoming and outgoing traffic. Employees, visitors and trucks that have access approval are screened at the main building. A smaller security station has been placed at the secondary entrance to the site. Vehicle traffic including common carriers, such as mail delivery trucks, are screened at this location.

3.3.1.10.2 Functional Areas and Major Components

The main and secondary Security Buildings are located at the entries to the site. They are classified as a New Business Occupancy area by the NFPA 101 (NFPA, 1997) and is classified as Type IIIN Unprotected Construction area by the New Mexico Building Code (NMBC, 1997) and as Type II Construction by NFPA 220 (NFPA, 1999). These buildings are designed to meet the occupant and exiting requirements set by the NFPA 101 (NFPA, 1997) and the construction type classifications set by the New Mexico Building Code (NMBC, 1997).

3.3.1.10.3 Building Construction

The Security Building superstructures are designed of structural steel framing.

The roof structures consist of metal decking over structural steel framing. The metal decking is covered with a built-up roof system. The roof assembly has a minimum combined thermal resistance value of R-20.

Exterior walls consist of a combination of architectural metal panels and glazing. The exterior wall assembly has a minimum combined thermal resistance value of R-10. The interior side of the exterior wall is faced with 16 mm (5/8 in) gypsum wallboard.

Interior non-load bearing walls are constructed of 92 mm (4 in) metal studs filled with batt insulation and faced with 16 mm (5/8 in) gypsum wallboard. Walls extend to 150 mm (6 in) above the ceiling or to the underside of the structure where required.

Floors in the Security Buildings consist of sealed concrete.

3.3.2 Structural Design Criteria

The structural and mechanical design load criteria are based on the environmental and geologic features of the National Enrichment Facility site identified in Section 3.2, Site Description, and the data presented in the accepted Industry Codes and Standards. The design criteria meets the applicable baseline design criteria established in 10 CFR 70.64, Requirements for new facilities or new processes at existing facilities (CFR, 2003). The design is based on the codes and loads discussed below.

As part of the Integrated Safety Analysis for external events, the following structures (buildings and areas) were determined to be safety significant and are required to withstand the design basis natural phenomena hazards and external hazards defined in Section 3.2:

- Separations Building Modules (UF_6 Handling Area, Process Services Area, and Cascade Halls)
- Blending and Liquid Sampling Area
- Cylinder Receipt and Dispatch Building
- TSB
- Centrifuge Test Facility.

A. Safety significant structures shall be designed to withstand the effects of external events (i.e., seismic, tornado and high winds, tornado missiles, snow and ice load, and maximum local precipitation) reflected in Section 3.2.

B. The UF_6 Handling Area, Cascade Hall, Blending and Liquid Sampling Area, and Ventilated Room shall be designed and maintained such that leakage is maintained within the values determined in Integrated Safety Analysis (ISA) consequence calculations.

C. The UBC Storage Pad shall be designed to preclude flooding due to maximum local precipitation reflected in Section 3.2.

D. Above ground liquid storage tanks and water impoundments shall be designed such that they do not pose a flooding risk that could damage critical structures and/or systems under an assumed catastrophic failure and release of full contents (may be shown either by design, amount of contents or physical location).

Items relied on for safety (IROFS) associated with facility structures are listed in Section 3.8, IROFS.

3.3.2.1 Codes and Standards

The following codes and standards are generally applicable to the structural design of the National Enrichment Facility:

- New Mexico Building Code (NMBC, 1997)
- Uniform Building Code (UBC, 1997)
- ASCE 7-98, Minimum Design Loads for Buildings and Other Structures (ASCE, 1998)
- ACI 318-99, Building Code Requirements for Structural Concrete (ACI, 1999)
- ACI 349-90, Code Requirements for Nuclear Safety Related Concrete Structures (ACI, 1990b)
- AISC Manual of Steel Construction, Ninth Edition (AISC, 1989)
- PCI Design Handbook, Fifth Edition (PCI, 1999)
- American Society of Testing and Materials (ASTM).

3.3.2.2 Structural Design Loads

3.3.2.2.1 Wind Loadings

The determination of wind pressure loadings and the design for wind loads for all safety significant structures and components exposed to wind are based on the requirements of ASCE 7-98 (ASCE, 1998). The determination of wind pressure loadings and the design for wind loads for all other structures and components exposed to wind are based on the requirements of the Uniform Building Code (UBC, 1997), Chapter 16 which further refers to the wind design requirements of ASCE 7-98, Section 6.0 (ASCE, 1998). The design wind for structures having no safety significance is based on a 50-year period of recurrence. The basic wind speed is 130 km/hr (80 mi/hr). The wind speed is based on an Exposure C category which is for open terrain with scattered obstruction areas as given in the Uniform Building Code (UBC, 1997). For structures that are safety significant, the design wind speed is 252 km/hr (157 mi/hr). This wind speed is based on a 100,000-year period of recurrence. All buildings on the NEF site are less than 18.2 m (60 ft) in height.

The design wind pressures and forces on the total building area calculated in accordance with procedures outlined in Section 6.4.2 of ASCE 7-98 (ASCE, 1998). The wind pressures acting on the main wind-force resisting systems are determined using the following formulas:

Velocity Pressure $q_z = 0.00256 K_z K_{zt} K_d V^2 I$ (lb/ft^2) (Eq. 3.3-1)

Design Pressure $p = qGC_p - q_i(GC_{pi})$ (lb/ft^2) (Eq. 3.3-2)

Where:

q_z = velocity pressure evaluated at height z above ground, psf

K_z = velocity pressure exposure coefficient evaluated at height z

K_{zt} = topographic factor

K_d = wind directionality factor

V = basic wind speed, mi/hr (corresponds to a 3-second gust speed at 10.1 m (33 ft) in exposure category C)

I = importance factor = 1.00. Safety significant structures have an increased safety factor due to design probability of 1.0E-5 of wind

p = design wind pressure, lb/ft^2

G = gust effect factor

C_p = external pressure coefficient

q_i = velocity pressure for internal pressure determination

GC_{pi} = product of internal pressure coefficient and gust factor

The design of wind pressures and forces on building components and cladding are calculated in accordance with procedures outlined in Section 6.5.12.4 of ASCE 7-98 (ASCE, 1998). Wind pressures on building components and cladding are determined using the following formula:

$p = q_h[(GC_p) - (GC_{pi})]$ (lb/ft^2) (Eq. 3.3-3)

Where:

p = design wind pressure, lb/ft^2

q_h = velocity pressure at roof height z = h (mean roof height), lb/ft^2

G = gust effect factor

C_p = external pressure coefficient

GC_{pi} = product of internal pressure coefficient and gust factor

The design wind pressure on other structures is calculated in accordance with procedures outlines in Chapter 16, Division III of the Uniform Building Code (UBC, 1997). The design wind pressure is determined using the following formula:

Design Pressure $P = C_e C_q q_s I_w$ (lb/ft^2) (Eq. 3.3-4)

Where:

C_e = combined height, exposure and gust factor coefficient from Table 16-G

C_q = pressure coefficient from Table 16-H

q_s = wind stagnation pressure at standard height of 10 m (33 ft)

I_w = wind importance factor from Table 16-K Occupancy Category

The design wind pressures and forces on the total building area calculated in accordance with procedures outlined in Section 1621.3 of the Uniform Building Code (UBC, 1997). The design of wind pressures and forces on building components and cladding are calculated in accordance with procedures outlined in Section 1622 of the Uniform Building Code (UBC,1997).

3.3.2.2.2 Cyclonic Loadings

3.3.2.2.2.1 Tornado

The safety significant structures and components exposed to wind are designed to withstand tornado loadings including tornado-generated missiles. The tornado parameters are based on a 100,000-year period of recurrence.

The design parameters applicable to the design tornado are as follows:

Design wind speed:	302 km/hr	(188 mi/hr)
Radius of damaging winds:	130 m	(425 ft)
Atmospheric pressure change (APC):	-390 kg/m^2	(-80 lb/ft^2)
Rate of APC:	-146 $kg/m^2/s$	(-30 $lb/ft^2/s$)

The wind pressures are determined and applied to the structures and buildings in the same manner as the wind loads described in Section 3.3.2.2.1, Wind Loadings. Internal pressure differential due to atmospheric pressure change is considered. The procedures used for transforming the impactive missile loadings into effective loads are discussed in Section 3.3.2.2.3, Projectile Protection.

3.3.2.2.2.2 Hurricane

The NEF site is approximately 805 km (500 mi) inland from the nearest coastline. Hurricane wind is not a governing condition in comparison to normal wind and tornado wind.

3.3.2.2.3 Projectile Protection

Projectile protection is provided for all equipment, systems and components in the safety significant areas such that internally generated or externally generated missiles will not cause the release of radioactive materials or prevent the safe and orderly shutdown of the facility.

3.3.2.2.3.1 Internal Projectiles

Internally generated projectiles are not a concern in the Separations Building. The types of equipment that are potential sources of projectiles are blowers, fans, pumps, compressors, high pressure gas cylinders and the centrifuges. The centrifuges have been tested to mechanical failure. These tests have demonstrated that the centrifuge casing will contain any internal projectiles generated as a result of a centrifuge failure. Likewise, in the Separations Building and other safety significant areas of the facility, the components of the other pieces of rotating equipment located in these areas that could become missiles do not have sufficient energy to break through their respective housings or casings. Also, there are no high energy piping systems in these areas that could be the source of jet impingements or pipe whip. High pressure gas cylinders will be handled and stored on site to preclude the generation of internal missiles.

3.3.2.2.3.2 External Projectiles

The only external projectiles that have been identified as a design consideration are tornado-generated missiles. The barriers and buildings protecting equipment and components in the safety significant areas are designed to withstand and absorb tornado generated missile impact loads without causing any damage to the protected equipment and components.

Aircraft crashes are not credible events for the NEF site. Additional information concerning aircraft crashes is found in Section 3.2.

A. Tornado-Generated Missiles

The tornado-generated missiles are associated with the tornado event described in Section 3.3.2.2.2.1, Tornado. The types of missiles selected and the related design parameters were determined as part of the tornado study for the NEF site. These missiles are associated with the design basis tornado (DBT), which has an annual probability of occurrence of 1.0E-5. The design parameters include:

<u>Missile: 2 in. x 4 in. timber plank, 6.80 kg (15 lb)</u>

Horizontal speed	137 km/hr	(85 mi/hr)
Maximum height above ground.	60 m	(200 ft)
Vertical speed	88 km/hr	(57 mi/hr)

<u>Missile: 76.2 mm (3 in) diameter, steel pipe, 34 kg (75 lb)</u>

Horizontal speed	80 km/hr	(50 mi/hr)
Maximum height above ground	9.1 m	(30 ft)
Vertical speed	48 km/hr	(30 mi/hr)

Missile: Automobile, 1361 kg (3000 lb)

Horizontal speed	32 km/hr	(20 mi/hr)

The missile impact generates two types of effects on the barriers and buildings. First are the local effects, and second are the overall responses of the barrier and portions thereof to missile impact. The procedures employed in the design of the barriers for those effects are described below.

B. Local Effects of Tornado-Generated Missiles on Building Structures

The missiles are categorized as either hard or soft relative to the target. A missile is considered hard if the average crushing or buckling limit stress of the missile is greater than the average contact stress required to cause local crushing and penetration of the target. Missiles not meeting the above condition are considered soft missiles. The timber missile is considered soft and the steel pipe missile is considered hard. For reinforced concrete targets, the formulas used to establish the missile depth of penetration (x) and scabbing thickness (t_s) are based on the Modified National Defense Research Committee Formula (NDRC) (ASCE, 1980) and the Army Corps of Engineers Formula (ACE) (ASCE, 1980) respectively.

The modified NDRC formulas for penetration is given by:

$$x = \sqrt{4KNWd\left(\frac{V}{1{,}000d}\right)^{1.80}} \quad , \text{for } \frac{x}{d} \le 2.0 \qquad \text{(Eq. 3.3-5)}$$

$$x = \left\{\left[KNW\left(\frac{V}{1000d}\right)^{1.80}\right] + d\right\} \quad , \text{for } \frac{x}{d} > 2.0 \qquad \text{(Eq. 3.3-6)}$$

The ACE Formula for scabbing is given by:

$$\frac{t_s}{d} = 2.12 + 1.36\frac{x}{d}, \text{ for } 0.65 \le \frac{x}{d} \le 11.75 \qquad \text{(Eq. 3.3-7)}$$

The variables used in the NDRC and ACE formulas are defined below:

N = missile shape factor which has a value of 0.72 for flat-nosed missiles

d = $\left(\frac{4A_c}{\pi}\right)^{\frac{1}{2}}$ = effective missile diameter, in.

W = missile weight, lbs.

K = $\frac{180}{\sqrt{f'c}}$

f'_c = ultimate compressive strength of concrete, psi

A_c = missile contact area, sq in.

x = missile depth of penetration, in.

t_s = scabbing threshold thickness, in.

V = striking velocity of missile, fps

Per Section C.7.2.2 of ACI 349-90 (ACI, 1990b), the concrete thickness required to resist hard missiles shall be at least 1.2 times the scabbing thickness, t_s. References indicate that the soft missiles will cause no local penetration with the exception of possible punching shear failure. Punching shear is calculated and checked against the requirements of ACI 349-90 (ACI, 1990b), Section C.7.2.3.

For steel targets, the formula used to establish the perforation thickness is the Ballistic Research Laboratory (BRL) Formula (ASCE, 1980).

The BRL Formula to determine the target thickness is given by:

$$\left(\frac{e}{d}\right)^{1.5} = \frac{DV^2}{1{,}120{,}000K_s^2} \qquad \text{(Eq. 3.3-8)}$$

Where:

K_s = Steel penetrability constant depending upon the grade of the steel target, usually taken as 1.0.

$D = \frac{W}{d^3}$ = missile caliber density, lbs/in^3

$d = \left(\frac{4A_c}{\pi}\right)^{\frac{1}{2}}$ = effective missile diameter, in.

A_c = missile contact area, sq in.

e = perforation thickness, in.

V = striking velocity of missile, fps

W = missile weight, lbs

References indicate that the recommended steel target thickness is 1.25 times the perforation thickness (ASCE, 1980, p. 346).

C. Overall Structural Response

In addition to local impact effects, the barriers and building structures are designed to resist the overall effects of missile impact. Various methods for designing to resist the overall effects of missile impact are available. In addition to the procedure outlined below, the different formulations as presented in ACI 349-90 (ACI, 1990b) may be used.

The response of a structure to missile impact depends largely on the location of impact, the dynamic properties of the structure (target), and the kinetic energy of the missile. For tornado-generated missiles, the assumption of a plastic collision between the missile and target is used where all of the missile momentum is transferred into the target. Based on this assumption, and that the target has elasto-plastic behavior, expressions for an equivalent static load concentrated at the impact area can be determined (ASCE, 1980). This load, in combination with other design loads, is evaluated using conventional design methods.

3.3.2.2.4 Water Level

Based on setting the grade level of the facility above the maximum foreseeable flood level, the only potential flooding of the facility results from local intense rainfall. Protection against flooding is provided by establishing the facility floor level at 0.15 m (0.5 ft) above the high point of finished grade elevation and all roads are set at least 0.45 m (1.5 ft) below this. In addition, in order to prevent general site flooding from the contributory areas above the site, an earth berm and intercept trench will be constructed uphill of the buildings. Based on these design features, the probability of the water level reaching the building finished floor is negligible. Section 3.2, provides in detail the effects of flood from local intense precipitation.

3.3.2.2.5 Seismic Loadings

3.3.2.2.5.1 Building Code Earthquake

All buildings and structures, including such items as equipment supports, are designed to withstand the earthquake loads defined in Chapter 16, Division IV of the Uniform Building Code (UBC, 1997). Every structure is designed to resist the total lateral seismic forces acting nonconcurrently in the direction of each of the main axes of the structure. Based on Figure 16-2, Seismic Zone Map of the United States, the NEF site is located in seismic zone 1.

Although much of the facility is of a critical nature, the additional safety factor for developing seismic forces for these structures is provided by using the occurrence probability of 10^{-4}. Based on this, all buildings will be taken as standard occupancy structures.

The seismic total design base shear in a given direction is determined by the following:

$$V = \frac{C_v I}{RT} W \qquad \text{(Eq. 3.3-9)}$$

The total design base shear need not exceed:

$$V = \frac{2.5 C_a I}{R} W \qquad \text{(Eq. 3.3-10)}$$

The total design base shear shall not be less than:

$$V = 0.11 C_a I W \qquad \text{(Eq. 3.3-11)}$$

Where:

V = Total design lateral force or base shear

C_a = Seismic coefficient, as set forth in Table 16-Q of the Uniform Building Code (UBC, 1997)

C_v = Seismic coefficient, as set forth in Table 16-R of the Uniform Building Code (UBC, 1997)

R = Numerical coefficient representative of the inherent overstrength and global ductility capacity of lateral-force-resisting systems as set forth in Table 16-N or 16-P of the Uniform Building Code (UBC, 1997)

I = Importance factor, as set forth in Table 16-K of the Uniform Building Code (UBC, 1997)

T = Elastic fundamental period of vibration, seconds

W = Total seismic dead load defined in Section 1630.1.1 of the Uniform Building Code (UBC, 1997)

3.3.2.2.5.2 Design Basis Earthquake

The Design Basis Earthquake (DBE) for the NEF site has a peak horizontal acceleration of 0.15g and peak vertical acceleration of 0.15g. These values correspond to a design basis earthquake with a return period of 10,000 years (1.0E-4 annual probability). The ultimate target performance goal is an annual probability of 1.0E-5. The difference between design and target performance is accounted for in the design process by confirmatory calculations (design will based on code allowables and safety factors, additional calculations will show that although these allowables are exceeded for the target performance goal, the ultimate capabilities will not be exceeded). For licensing purposes, soil amplification factors are based on Soil Class C. This assumption will be verified during final design. Refer to Section 3.2, for a detailed discussion of the geology and seismicity of the region used in determining the DBE.

3.3.2.2.6 Precipitation Loadings

3.3.2.2.6.1 Snow Loadings

Snow loadings on roofs and other exposed surfaces for non-safety significant structures are determined in accordance with the Uniform Building Code (UBC, 1997), Chapter 16, Division II. The design parameters identified below are based on a mean return period of 50 years.

Snow loadings on roofs of safety significant buildings are based on a Ground Snow Load (p_a) of 156 kg/m^2 (32 lb/ft^2). Further discussion for the basis of this load can be found in Section 3.2. All other parameters and determination of snow drifts will be the same as the non-safety significant structures.

3.3.2.2.6.2 Rainfall Loadings

Rainfall loadings on roofs and other exposed surfaces result from two different events. The first event is normal heavy rainfall having a 100 year return period. Loads on the roof occur during this event as a result of assuming that the primary roof drains are blocked. The load equals the depth of water required before water can flow out of the secondary roof drains. The roof drainage systems (including secondary roof drains) will be designed such that the amount of rainfall that can collect on the roof does not exceed the normal roof design live load.

The second event is localized intense rainfall. Refer to Section 3.2.3.4.4 for further discussion. The load equals the depth of water that accumulates in excess of the roof drains capacity. This is used for the design of the safety significant areas only.

3.3.2.2.7 Process and Equipment Derived Loadings

The various buildings and structures are designed to support the equipment, piping, duct and tray associated with them. Dead loads, fluid loads, impact loads, seismic loads and other dynamic loads are accounted for in the design. In addition to the buildings, individual supports are designed to withstand these same types of loads.

3.3.2.2.7.1 Equipment Loads

All pieces of equipment that exceed 454 kg (1,000 lb) dead weight, including contents, are accounted for individually in the design. The remaining equipment is accounted for in the building design by including an appropriate uniform dead load for a particular area.

3.3.2.2.7.2 Piping Loads

Piping loads transmitted through pipe racks to the building are based on combined dead and live loads of 244 kg/m^2 (50 lb/ft^2) of pipe run area for each pipe rack level. The area considered is the length times the width of the pipe runs.

3.3.2.2.7.3 HVAC Loads

HVAC duct loads transmitted through supports to the building are based on combined dead and live loads of 146 kg/m^2 (30 lb/ft^2) of duct run area. The area considered is the length times the width of the HVAC duct runs.

3.3.2.2.7.4 Electrical Tray and Conduit Loads

Electrical tray and conduit loads transmitted through supports and electrical racks to the building are based on combined dead and live loads of 74 kg/m (50 lb/ft) of tray and a 91 kg (200 lb) concentrated load at mid-span of the tray and 30 kg/m (20 lb/ft) of conduit.

3.3.2.2.8 Combined Loadings for Structures

Load combinations for concrete structures and components for the safety significant structures are based on ACI 349-90 (ACI, 1990b). These combinations are listed in Section 3.3.2.2.8.3.1. Load combinations for other concrete structures are based on ASCE 7-98 (ASCE, 1998). These combinations are listed in Section 3.3.2.2.8.3.2. All concrete structures are designed using the ACI Strength Design Method (ACI, 1999). Load combinations for steel structures and components for all buildings are based on ASCE 7-98 (ASCE, 1998). These load combinations are listed in Section 3.3.2.2.8.3.3. All structural steel is designed using the AISC Allowable Stress Method (AISC, 1989). Loads are considered to act in various load combinations as listed in this section. Results are checked for whatever combination produces the most unfavorable effects for the buildings, foundations or other structural components being considered.

All major loads encountered and/or postulated in a safety significant structure or component are listed in three categories described below.

3.3.2.2.8.1 Normal Loads

Normal loads are those loads encountered during normal facility operation. They include the following:

A. Dead (D)

Dead loads include gravitational load of structures, permanent equipment, piping, static liquid, long term stored materials, permanent partitions and any other permanent static load.

B. Live (L or L_R)

Live loads include the weight of moveable objects such as personnel and equipment, temporarily stored materials, tools, moveable partitions, transporters, hoists and cranes. Design live loads, including impact loads, used are in accordance with Section 4.0 and Table 4-1 of ASCE 7 (SBCCI, 1999).

C. Self-Straining (T)

Self-straining forces and effects arise from the restraint of a structural member from expansion or contraction due to temperature change, shrinkage, creep or differential settlement.

D. Pressure (F)

Lateral and vertical pressure of liquid or gases due to their containment within a structure.

E. Lateral Earth Pressure (H)

The lateral earth pressure acting on foundations, buried walls or retaining walls.

F. Environmental Loads

Environmental loads include the following:

1. Snow (S)
   Snow loads are discussed in Section 3.3.2.2.6, Precipitation Loadings.
2. Rainfall (R)
   Normal rainfall loads are discussed in Section 3.3.2.2.6.
3. Wind (W)
   Wind loads are discussed in Section 3.3.2.2.1, Wind Loadings.
4. Earthquake (E_o)

   Building code earthquake loads are discussed in Section 3.3.2.2.5, Seismic Loadings.

G. Process and Equipment Reactions (R_o)

Process and equipment derived loads are discussed in Section 3.3.2.2.7, Process and Equipment Derived Loadings.

H. Postulated Pipe Break Loads

1. Pressure Differential (Pa) - Differential pressure load generated by a postulated pipe break. Load to be determined during final design based on line size and maximum pressure.
2. Jet Impingement Load (Y_j) - Jet impingement load generated by a postulated pipe break. Load to be determined during final design based on line size and maximum pressure.
3. Missile Impact Load (Y_m) - Missile impact load, including pipe whip, generated by a postulated pipe break. Load to be determined during final design based on line size and maximum pressure.
4. Pipe Reaction (Y_r) - Load generated by broken pipe during postulated pipe break. Load to be determined during final design based on line size and maximum pressure.

3.3.2.2.8.2 Extreme Environmental Loads

Extreme environmental loads are those loads that are credible but highly improbable. They include the following:

A. Design Basis Tornado (W_t)

The Design Basis Tornado loads are made up of 3 load components acting in various combinations. The load components are:

1. Tornado wind velocity pressure (W_w)
2. Tornado induced differential pressure (W_p)
3. Tornado generated missile load (W_m)

Items 1. and 2. are discussed in Section 3.3.2.2.2. Item 3. is discussed in Section 3.3.2.2.3.

The three load components can act in the following combinations as described in ACI 349-90 (ACI, 1990b).

a. $W_t = W_w$
b. $W_t = W_p$
c. $W_t = W_m$
d. $W_t = W_w + W_m$
e. $W_t = W_w + 0.5\ W_p$
f. $W_t = W_w + 0.5\ W_p + W_m$

B. Safe Shutdown Earthquake (E_s)

Loads from the Safe Shutdown Earthquake (i.e., DBE) are discussed in Section 3.3.2.2.5.

C. Design Basis Flood (DBFL)

Loads from the Design Basis Flood are discussed in Section 3.3.2.2.4.

D. Truck and Gas Pipeline Hazards

Explosion hazards from trucks (e.g., propane trucks) on highways near the NEF site are described in Section 3.2.1.2.1. Explosion hazards from gas pipelines near the NEF site are described in Section 3.2.2.4, Industrial Areas. During detailed design of specific buildings and areas, pressure loads due to postulated truck and pipeline explosions will be considered. The pressure loads will be developed in accordance with the underlying assumptions used in the explosion hazard assessments described in Sections 3.2.1.2.1 and 3.2.2.4. These buildings and areas include: Separations Building Modules (UF_6 Handling Area, Process Services Area and Cascade Halls), Blending and Liquid Sampling Area, Cylinder Receipt and Dispatch Building, Technical Services Building and the Centrifuge Test Facility. As described in Section 3.3.1, Buildings and Major Components, these buildings and areas are constructed of concrete.

3.3.2.2.8.3 Combined Load Applications

The load combinations defined in this section are applied to all structures, components and equipment supports.

A. Load Combinations For Structures Combining Factored Loads Using Strength Design (Concrete)

All of the following load combinations shall be satisfied for concrete structures for the safety significant areas:

1. $U = 1.4D + 1.4F + 1.7(L_R \text{ or } S \text{ or } R) + 1.7H + 1.4R_o$
2. $U = 1.4D + 1.4F + 1.7L + 1.7H + 1.7Eo + 1.7Ro$
3. $U = 1.4D + 1.4F + 1.7L + 1.7H + 1.7W + 1.7Ro$
4. $U = D + F + L + H + T + Ra + 1.25Pa$
5. $U = D + F + L + H + T + Ra + 1.15Pa + 1.0(Yr + Yj + Ym) + 1.15Eo$
6. $U = 1.05D + 1.05F + 1.3L + 1.3H + 1.05T + 1.3Ro$
7. $U = 1.05D + 1.05F + 1.3L + 1.3H + 1.3Eo + 1.05T + 1.3Ro$
8. $U = 1.05D + 1.05F + 1.3L + 1.3H + 1.3W + 1.05T + 1.3Ro$

For extreme environmental conditions the following load combinations are satisfied:

9. $U = D + F + L + H + T + R_o + E_s$
10. $U = D + F + L + H + T + R_o + W_t$
11. $U = D + F + L + H + T + R_a + 1.0P_a + 1.0(Y_r + Y_j + Y_m) + 1.0E_s$
12. U - Used for concrete structures, U is the required strength to resist factored loads or related internal moments, shears and forces, based on methods described in ACI 318 (ACI, 1999).

B. Load Combinations For Structures Combining Factored Loads Using Strength Design (Concrete)

All of the following load combinations shall be satisfied for all concrete structures:

1. $U = 1.4(D + F)$
2. $U = 1.2(D + F + T) + 1.6(L+H) + 0.5(L_r \text{ or } S \text{ or } R)$

3. U = $1.2D + 1.6(L_r \text{ or } S \text{ or } R) + (0.5L \text{ or } 0.8W)$
4. U = $1.2D + 1.6W + 0.5L + 0.5(L_r \text{ or } S \text{ or } R)$
5. U = $1.2D + 1.0E_o + 0.5L + 0.2S$
6. U = $0.9D + 1.6W + 1.6H$
7. U = $0.9D + 1.0E_o + 1.6H$
8. U - Used for concrete structures, U is the required strength to resist factored loads or related internal moments, shears and forces, based on methods described in ACI 318-99 (ACI, 1999).

C. Load Combinations For Structures Combining Nominal Loads Using Allowable Stress Design (Steel)

All of the following combinations shall be satisfied for steel structures:

1. S = D
2. S = $D + L + F + H + T + (L_r \text{ or } S \text{ or } R)$
3. S = $D + (W \text{ or } 0.7E_o) + L + (L_r \text{ or } S \text{ or } R)$
4. S = $0.6D + W + H$
5. S = $0.6D + 0.7E_o + H$

For extreme environmental conditions the following load combinations are satisfied:

6. S = 0.625(D + L + T + Eo)
7. S = 0.625(D + L + T + Wt)
8. S - Used for structural steel, S is the required section strength based on the elastic design methods and the allowable stresses defined in the AISC Manual of Steel Construction-Allowable Stress Design (AISC, 1989).

Load Combinations and Requirements for Foundations

All foundations are checked against sliding and overturning due to earthquake, wind, Design Basis Earthquake and Design Basis Tornado in accordance with the following:

Minimum Factors of Safety

Load Combination	Overturning	Sliding
$D + H + E_o$	1.5	2.0
$D + H + W$	1.5	2.0
$D + H + E_s$	1.5	2.0
$D + H + W_t$	1.5	2.0

The allowable stresses cannot exceed 0.7 times the ultimate tensile strength ($0.7F_u$) in axial tension nor 0.7 times the ultimate tensile strength times the ratio of plastic section modulus to elastic section modulus ($0.7F_u$ Z/S).

3.3.2.3 Foundations

Foundations are shallow concrete spread footings. In areas where the footings bear on in situ rock, the allowable bearing pressure is 7,000 lb/ft^2. In areas where the footings bear in existing or new fill areas, the allowable bearing pressure is 3,000 lb/ft^2. The allowable bearing pressure may be higher in areas where the fill material is entirely rock.

3.3.3 References

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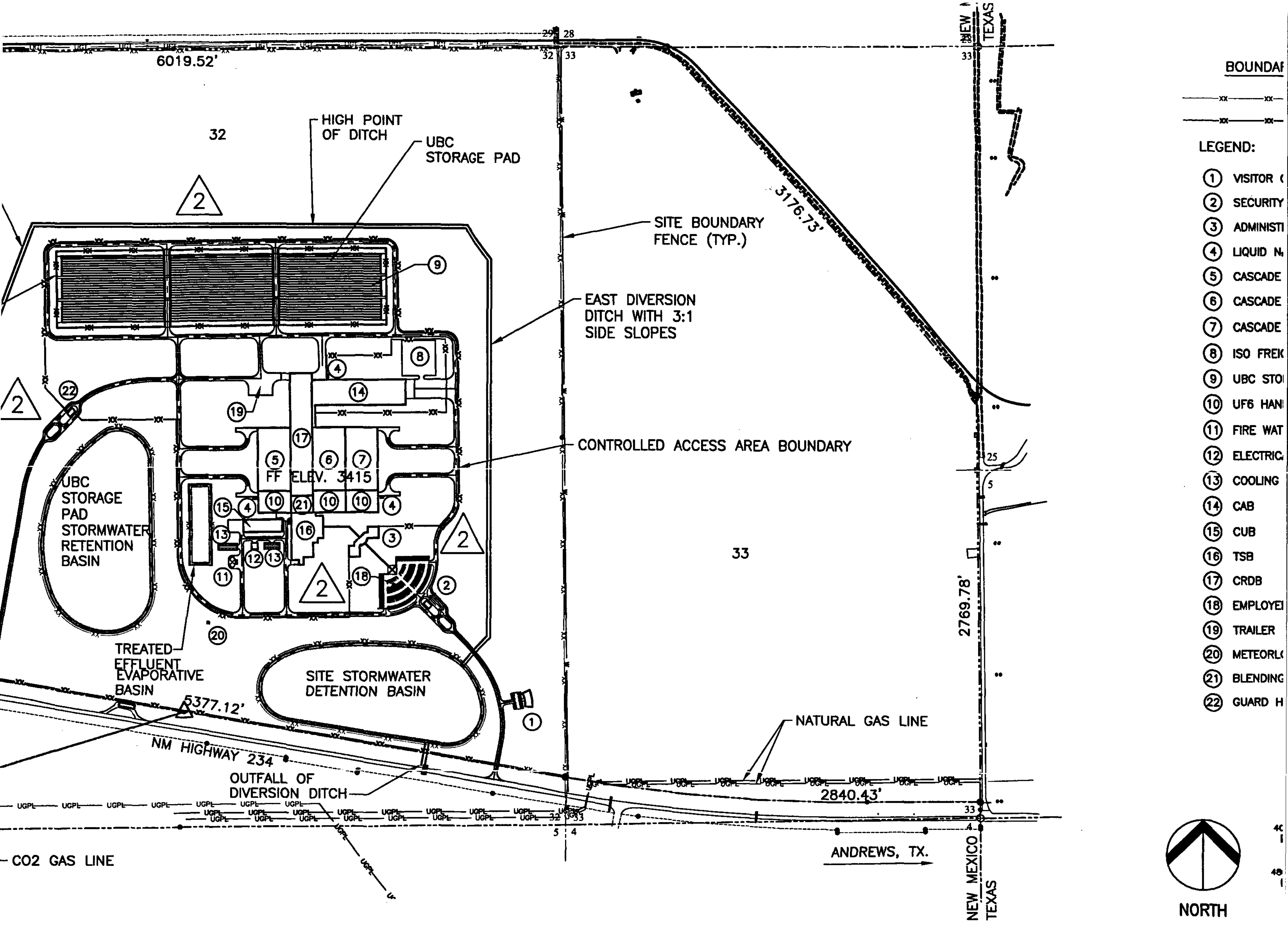
TABLES

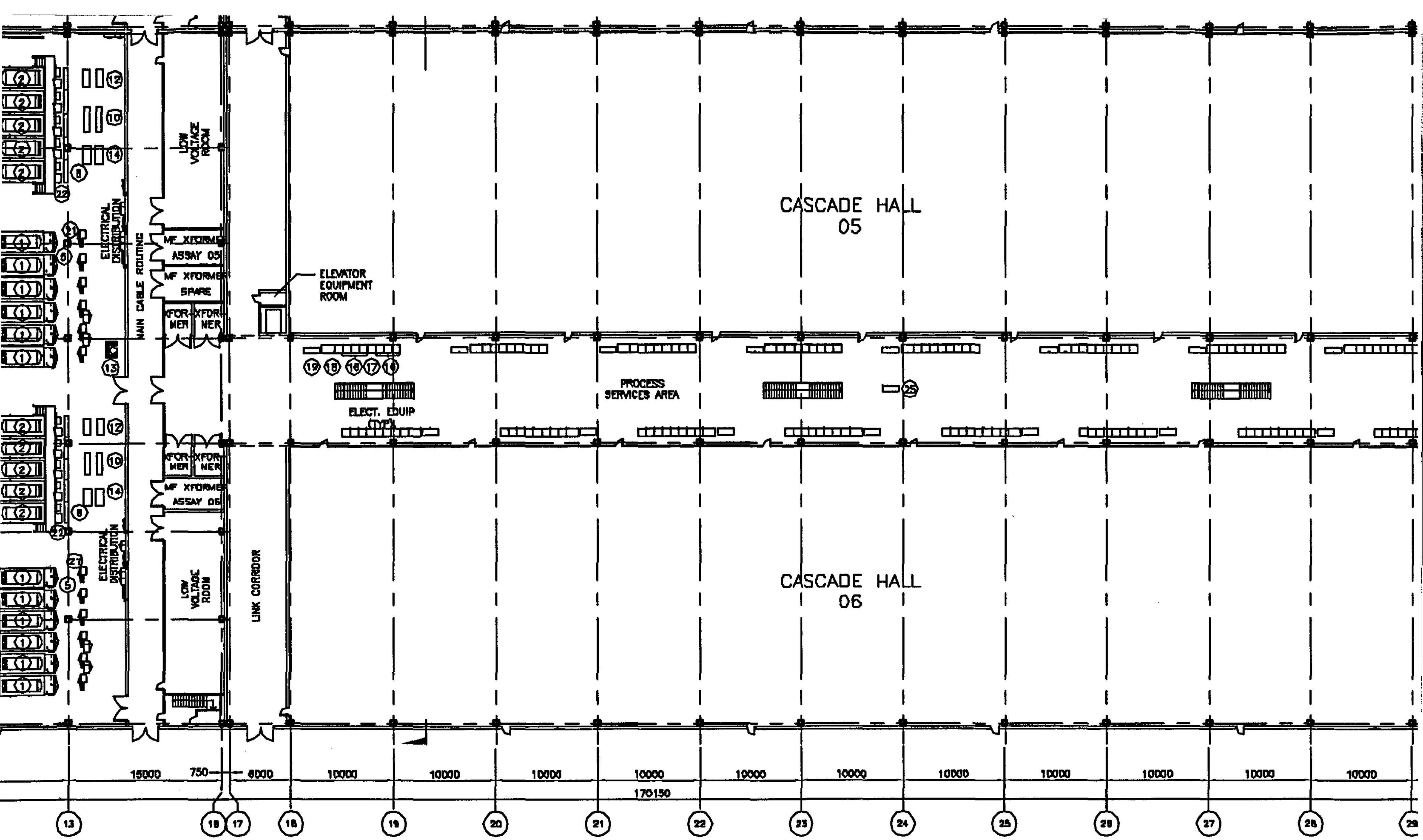
Table 3.3-1 Distances To Site Boundary and To Restricted Area Boundary and Wind Frequencies

Compass Direction from Facility	Distance from Facility to Site Boundary (meters)	(feet)	Distance from Facility Building Complex to Restricted Area Boundary (meters)	(feet)	Distance from UBC Storage Pad to Restricted Area Boundary (meters)	(feet)	Frequency of Wind (%)
S	417	1368	26.4	87	81.6	268	5.66
SSW	417	1368	26.4	87	-	-	3.98
SW	422	1384	28.8	94	-	-	4.91
WSW	503	1650	148.8	488	-	-	4.87
W	769	2522	168.0	551	33.6	110	6.29
WNW	1071	3513	168.0	551	-	-	5.52
NW	1072	3516	182.4	598	-	-	7.52
NNW	995	3264	93.6	307	-	-	10.80
N	995	3264	93.6	307	28.8	94	20.40
NNE	754	2473	93.6	307	-	-	7.35
NE	581	1906	100.8	331	-	-	5.46
ENE	540	1771	72.0	236	-	-	4.68
E	540	1771	57.6	189	33.6	110	4.45
ESE	540	1771	33.6	110	-	-	2.42
SE	487	1597	28.8	94	-	-	2.69
SSE	417	1368	26.4	87	-	-	3.04

FIGURES

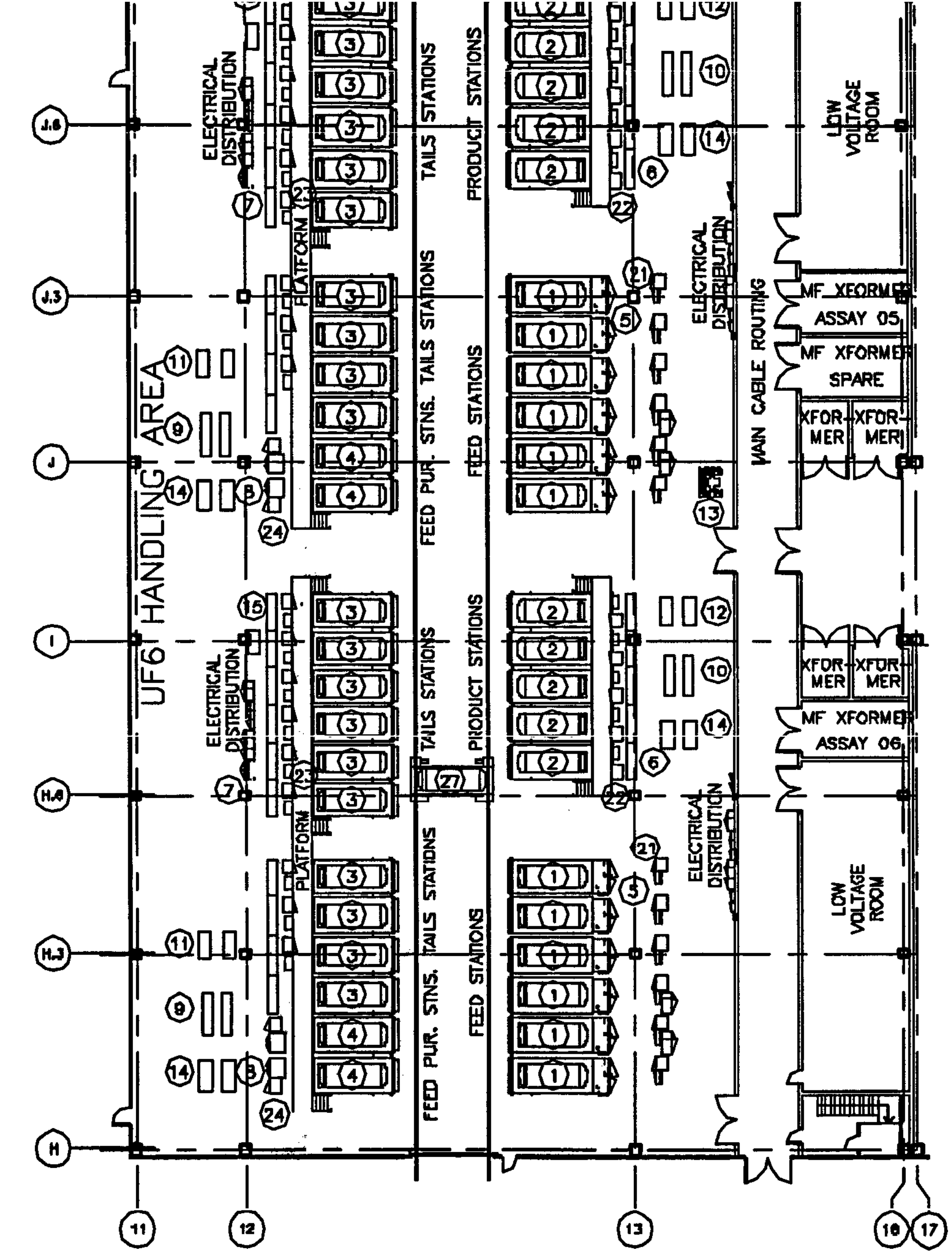


6019.52'
32
HIGH POINT OF DITCH
UBC STORAGE PAD
SITE BOUNDARY FENCE (TYP.)
3176.73'
EAST DIVERSION DITCH WITH 3:1 SIDE SLOPES
CONTROLLED ACCESS AREA BOUNDARY
FF ELEV. 3415
UBC STORAGE PAD STORMWATER RETENTION BASIN
TREATED EFFLUENT EVAPORATIVE BASIN
SITE STORMWATER DETENTION BASIN
5377.12'
NM HIGHWAY 234
OUTFALL OF DIVERSION DITCH
NATURAL GAS LINE
2840.43'
2769.78'
33
CO2 GAS LINE
ANDREWS, TX.
NEW MEXICO
TEXAS
NORTH
BOUNDAI
LEGEND:
1 VISITOR
2 SECURITY
3 ADMINISTI
4 LIQUID N
5 CASCADE
6 CASCADE
7 CASCADE
8 ISO FREI
9 UBC STO
10 UF6 HAN
11 FIRE WAT
12 ELECTRIC
13 COOLING
14 CAB
15 CUB
16 TSB
17 CRDB
18 EMPLOYEI
19 TRAILER
20 METEORL
21 BLENDING
22 GUARD H



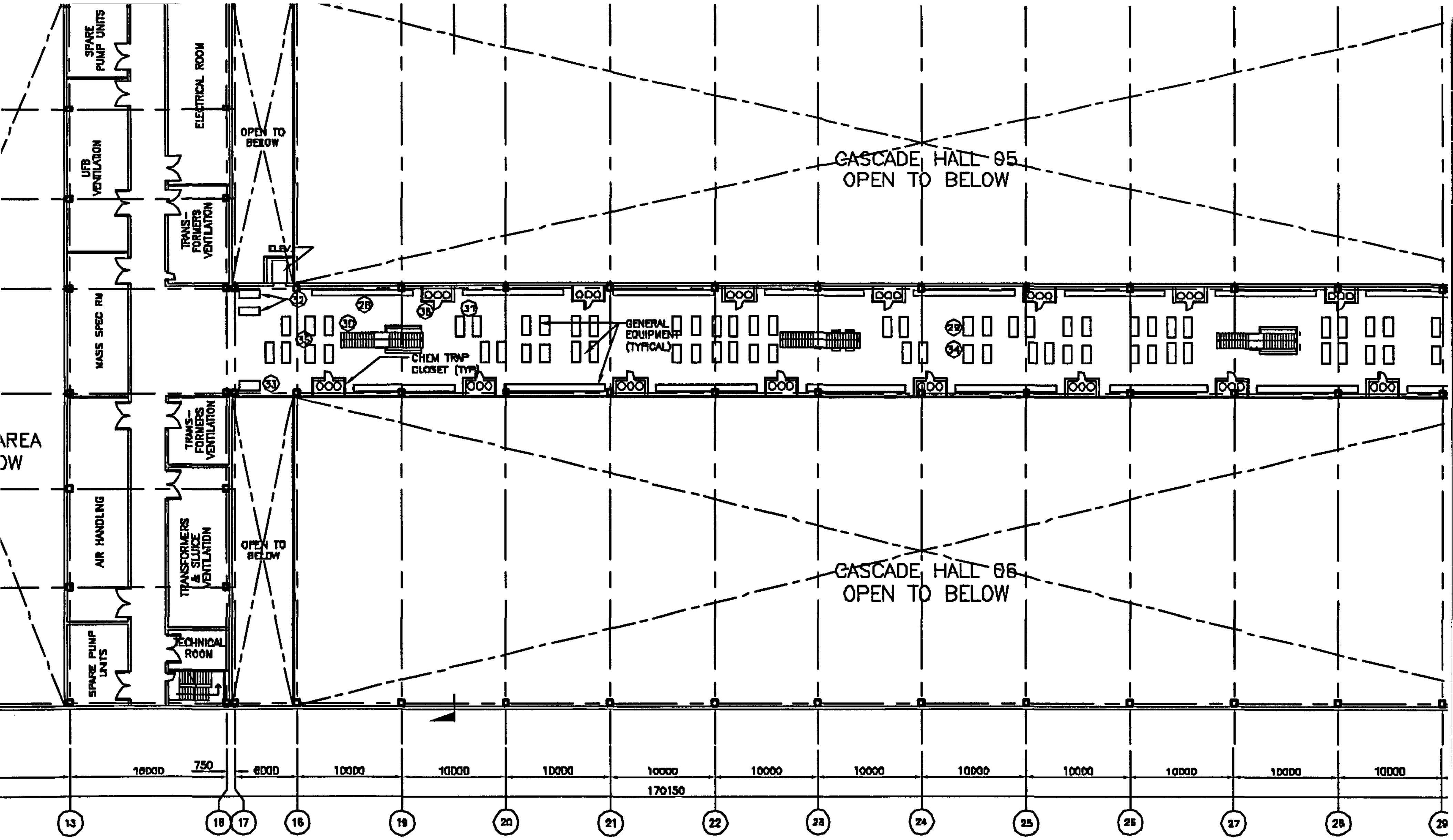
1,000,000 SWU MODULE

21	FEED STATION LOCAL CONTROL CENTER
22	PRODUCT STATION LOCAL CONTROL CENTER
23	TAILS STATION LOCAL CONTROL CENTER
24	FEED PURIFICATION STATION LOCAL CONTROL CENTER
25	PROCESS SERVICES AREA LOCAL CONTROL CENTER
26	*



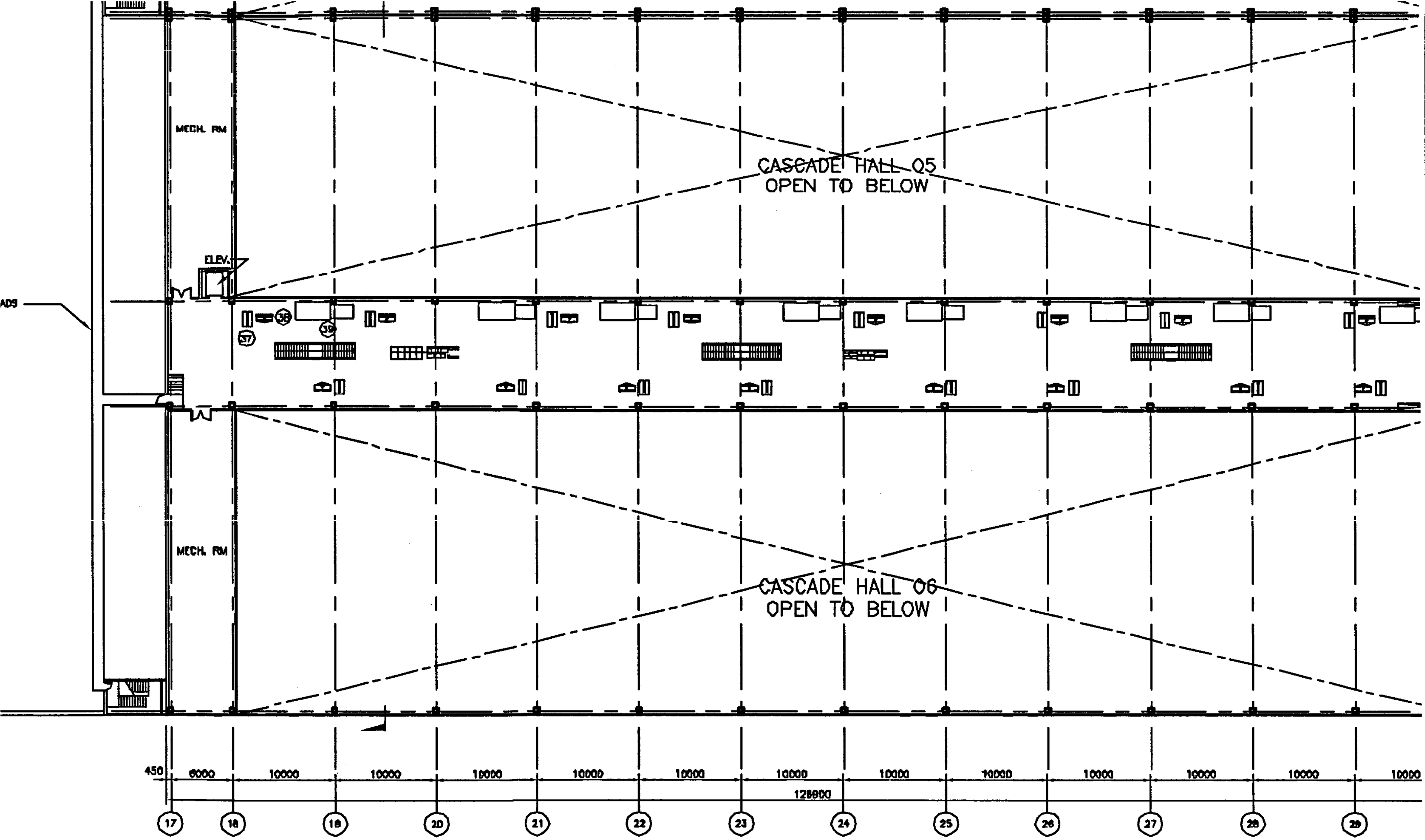
1,000,000 SWU MODULE

21 FEED STATION LOCAL CONTROL CENTER
22 PRODUCT STATION LOCAL CONTROL CENTER
23 TAILS STATION LOCAL CONTROL CENTER
24 FEED PURIFICATION STATION LOCAL CONTROL CENTER
25 PROCESS SERVICES AREA LOCAL CONTROL CENTER
26 *
27 RAIL TRANSPORTER
28 CASCADE VALVE STATION



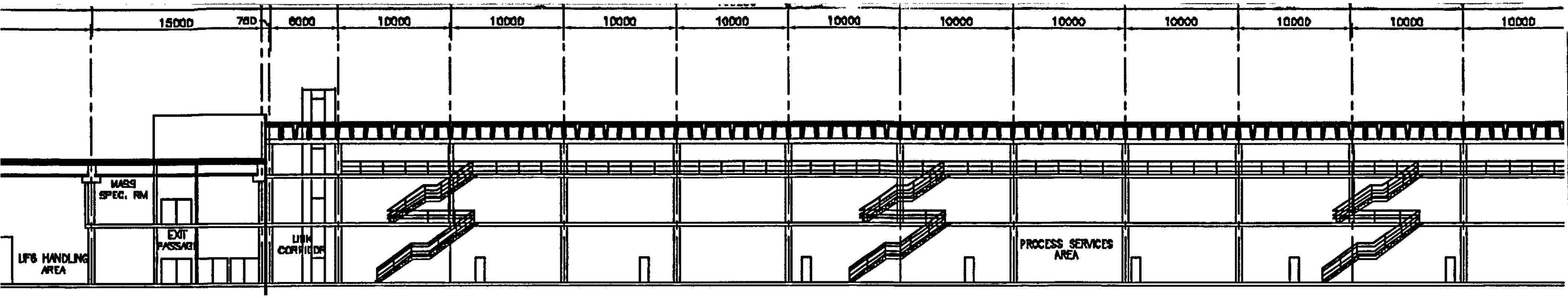
1,000,000 SWU MODULE

21 FEED STATION LOCAL CONTROL CENTER
22 PRODUCT STATION LOCAL CONTROL CENTER
23 TAILS STATION LOCAL CONTROL CENTER
24 FEED PURIFICATION STATION LOCAL CONTROL CENTER
25 PROCESS SERVICES AREA LOCAL CONTROL CENTER
26 *
27 RAIL TRANSPORTER
28 CASCADE VALVE STATION

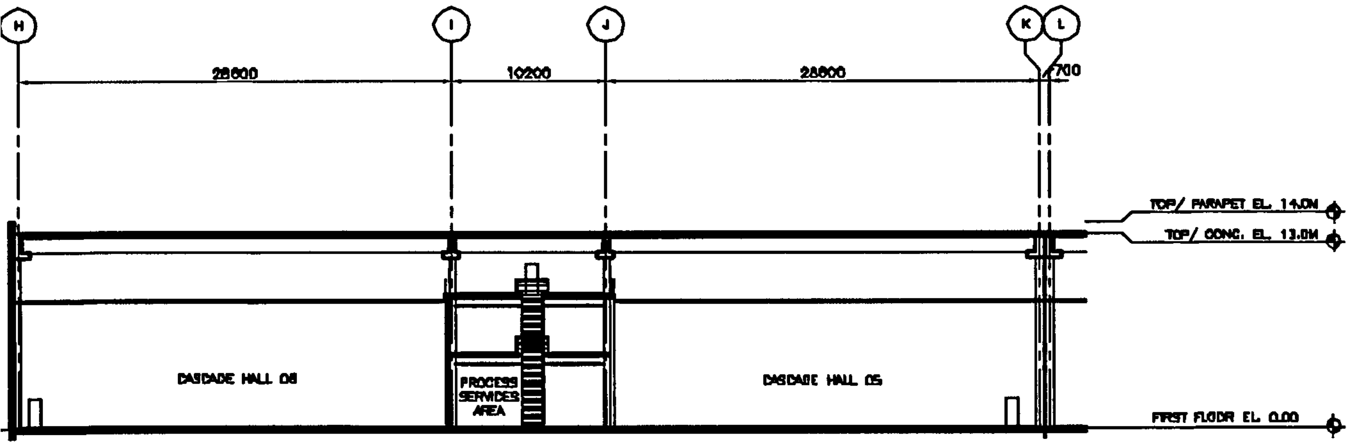


1,000,000 SWU MODULE

21 FEED STATION LOCAL CONTROL CENTER
22 PRODUCT STATION LOCAL CONTROL CENTER
23 TAILS STATION LOCAL CONTROL CENTER
24 FEED PURIFICATION STATION LOCAL CONTROL CENTER
25 PROCESS SERVICES AREA LOCAL CONTROL CENTER
26 *
27 RAIL TRANSPORTER
28 CASCADE VALVE STATION
29 PRODUCT PUMPING TRAIN



SECTION 1
FIGURE 3.3-5
FIGURE 3.3-4
FIGURE 3.3-2
FIGURE 3.3-6



SECTION 2
FIGURE 3.3-5
FIGURE 3.3-4
FIGURE 3.3-2
FIGURE 3.3-6

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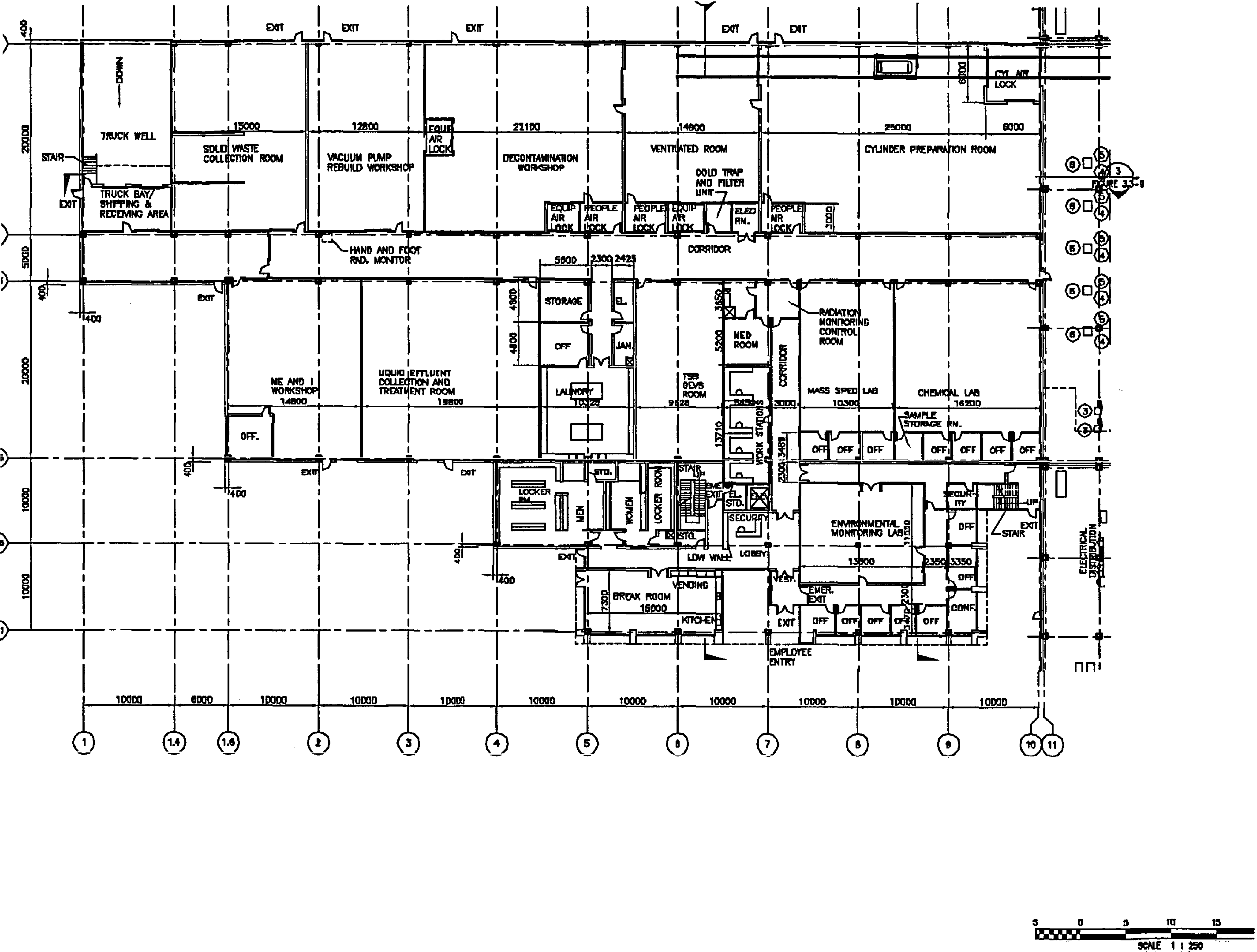
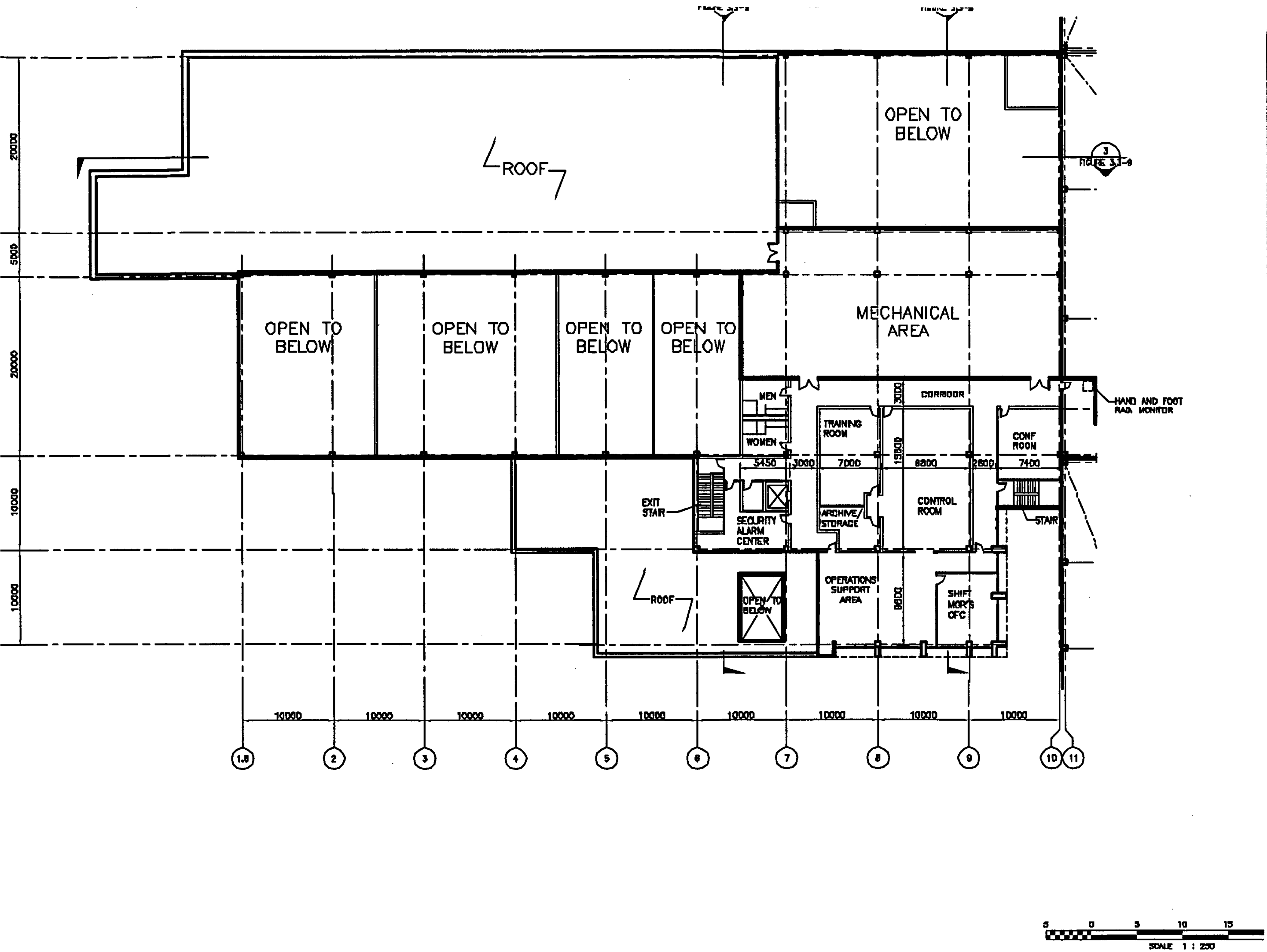

EXIT
TRUCK WELL
DOWN
STAIR
TRUCK BAY/ SHIPPING & RECEIVING AREA
SOLID WASTE COLLECTION ROOM
15000
VACUUM PUMP REBUILD WORKSHOP
12800
EQUIP AIR LOCK
DECONTAMINATION WORKSHOP
22100
VENTILATED ROOM
14800
COLD TRAP AND FILTER UNIT
CYLINDER PREPARATION ROOM
25000
6000
CYL AIR LOCK
EQUIP AIR LOCK
PEOPLE AIR LOCK
ELEC RM.
CORRIDOR
HAND AND FOOT RAD. MONITOR
FIGURE 3.3-9
STORAGE
EL.
OFF
JAN
LAUNDRY
10325
NE AND I WORKSHOP
14800
LIQUID EFFLUENT COLLECTION AND TREATMENT ROOM
19800
OFF.
TSB GEVS ROOM
NED ROOM
CORRIDOR
WORK STATION
RADIATION MONITORING CONTROL ROOM
MASS SPEC LAB
10300
CHEMICAL LAB
16200
SAMPLE STORAGE RM.
OFF
LOCKER RM.
MEN
WOMEN
LOCKER ROOM
STD.
STAIR
EMER. EXIT
STG.
SECURITY
LDW WALL
LOBBY
ENVIRONMENTAL MONITORING LAB
13800
SECURITY
STAIR
UP
CONF.
EMER. EXIT
VEST.
BREAK ROOM
15000
VENDING
KITCHEN
EMPLOYEE ENTRY
ELECTRICAL DISTRIBUTION
10000
6000
1
1.4
1.6
2
3
4
5
6
7
8
9
10
11
SCALE 1 : 250



ROOF
OPEN TO BELOW
OPEN TO BELOW
OPEN TO BELOW
OPEN TO BELOW
OPEN TO BELOW
MECHANICAL AREA
CORRIDOR
HAND AND FOOT RAD. MONITOR
MEN
WOMEN
TRAINING ROOM
CONF ROOM
CONTROL ROOM
EXIT STAIR
SECURITY ALARM CENTER
ARCHIVE/STORAGE
STAIR
ROOF
OPEN TO BELOW
OPERATIONS SUPPORT AREA
SHIFT MGR'S OFC
FIGURE 3.3-9
20000
5000
20000
10000
10000
3000
15600
9600
5450
3000
7000
8800
2800
7400
10000
10000
10000
10000
10000
10000
10000
10000
10000
1.8
2
3
4
5
6
7
8
9
10
11
5
0
5
10
15
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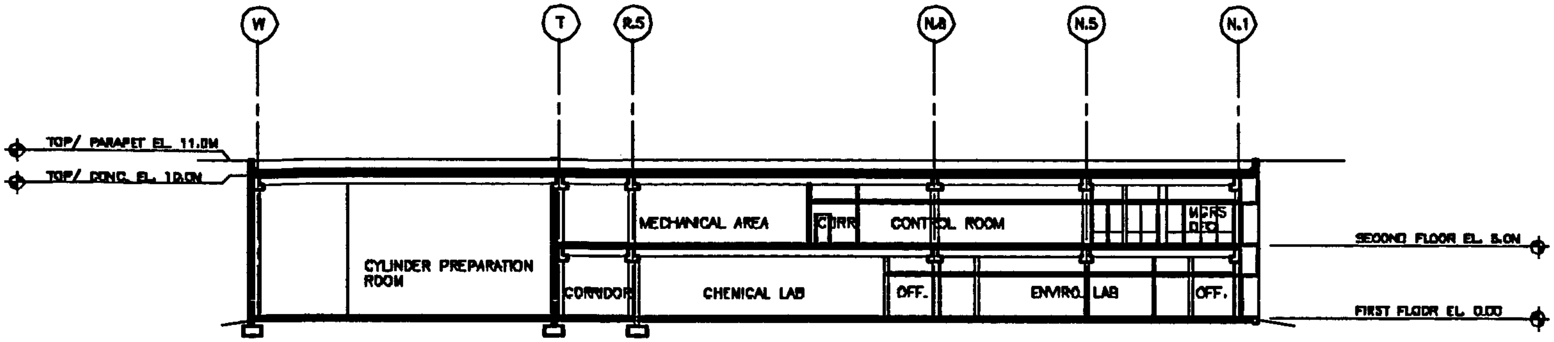
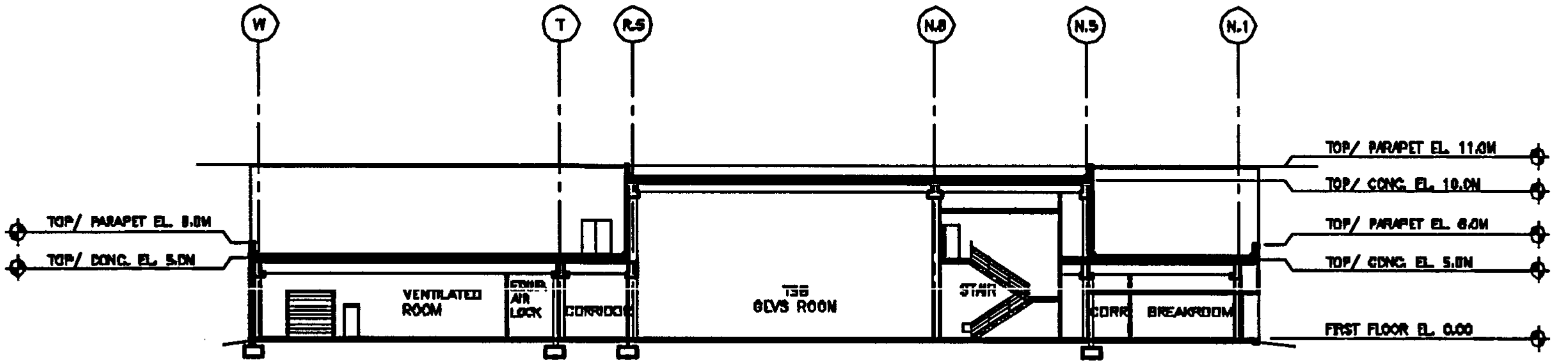
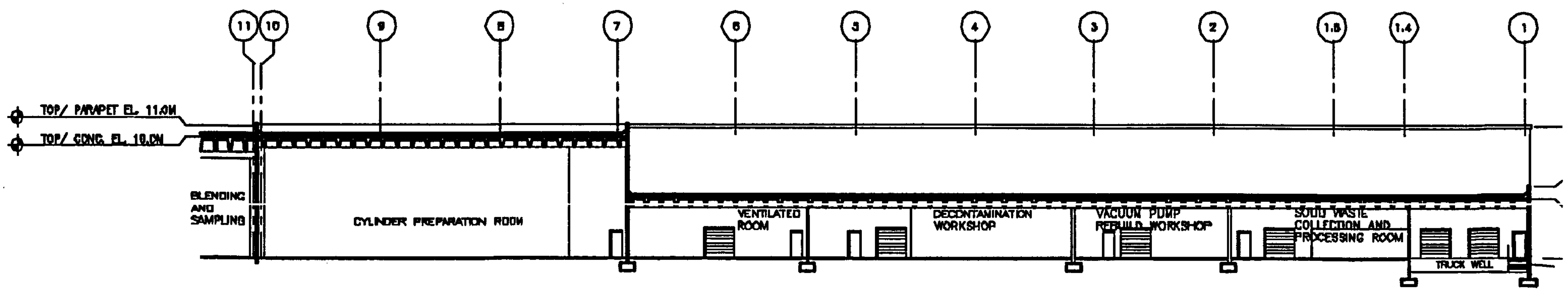


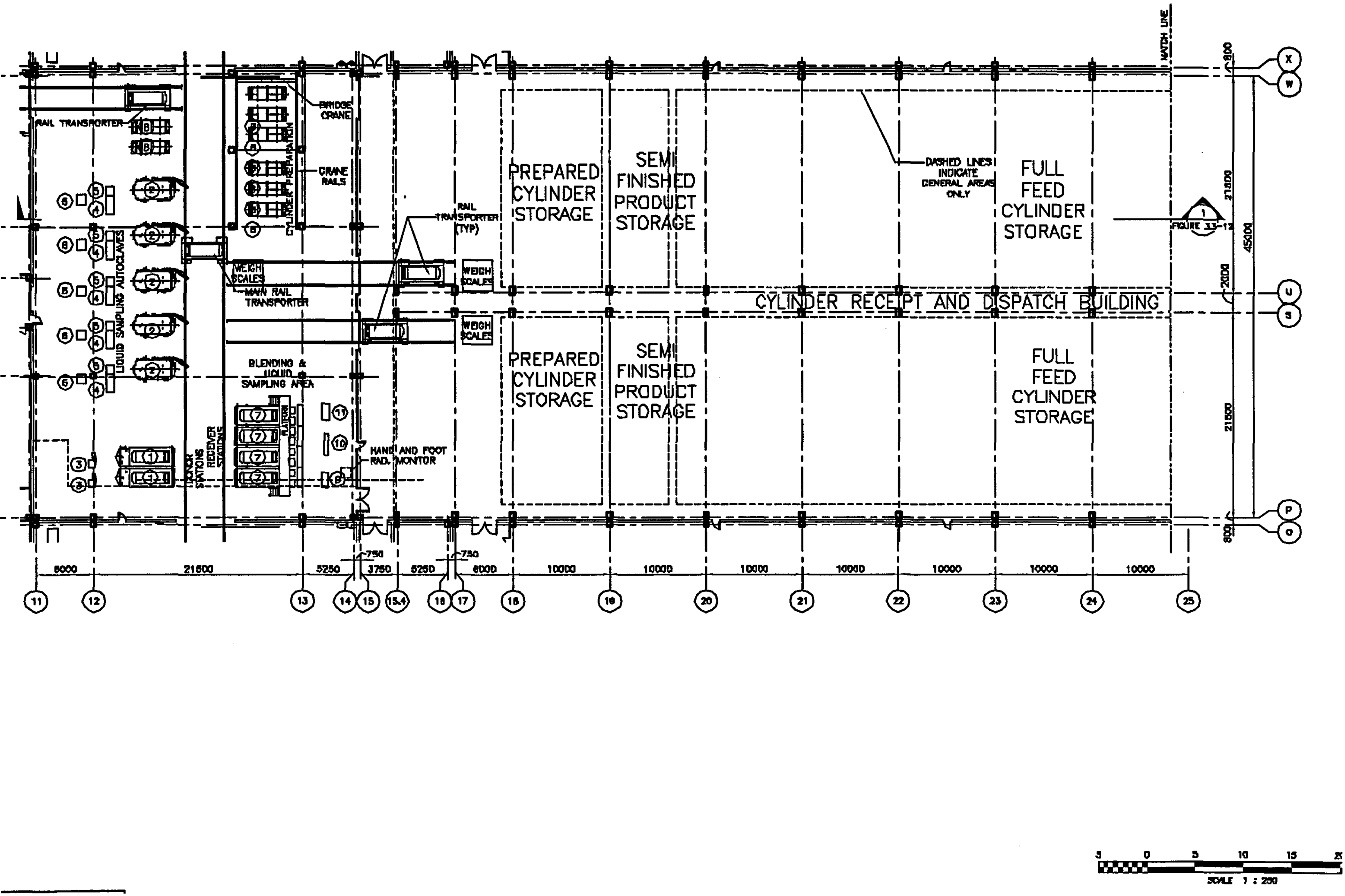
SECTION 1 FIGURE 3.3-8 FIGURE 3.3-7 FIGURE 3.3-9



SECTION 2 FIGURE 3.3-8 FIGURE 3.3-7 FIGURE 3.3-9



SECTION 3 FIGURE 3.3-8 FIGURE 3.3-7 FIGURE 3.3-9



MATCH LINE
RAIL TRANSPORTER
BRIDGE CRANE
CRANE RAILS
CYLINDER PREPARATION
LIQUID SAMPLING AUTOCLAVES
WEIGH SCALES
MAIN RAIL TRANSPORTER
RAIL TRANSPORTER (TYP)
WEIGH SCALES
WEIGH SCALES
PREPARED CYLINDER STORAGE
SEMI FINISHED PRODUCT STORAGE
DASHED LINES INDICATE GENERAL AREAS ONLY
FULL FEED CYLINDER STORAGE
FIGURE 3.3-12
CYLINDER RECEIPT AND DISPATCH BUILDING
BLENDING & LIQUID SAMPLING AREA
DONOR STATIONS
RECEIVER STATIONS
HAND AND FOOT RAD. MONITOR
PREPARED CYLINDER STORAGE
SEMI FINISHED PRODUCT STORAGE
FULL FEED CYLINDER STORAGE
6000 21500 5250 3750 5250 6000 10000 10000 10000 10000 10000 10000 10000
750
750
11 12 13 14 15 15.4 16 17 18 19 20 21 22 23 24 25
X W U S P O
800 21500 2000 21500 800 45000
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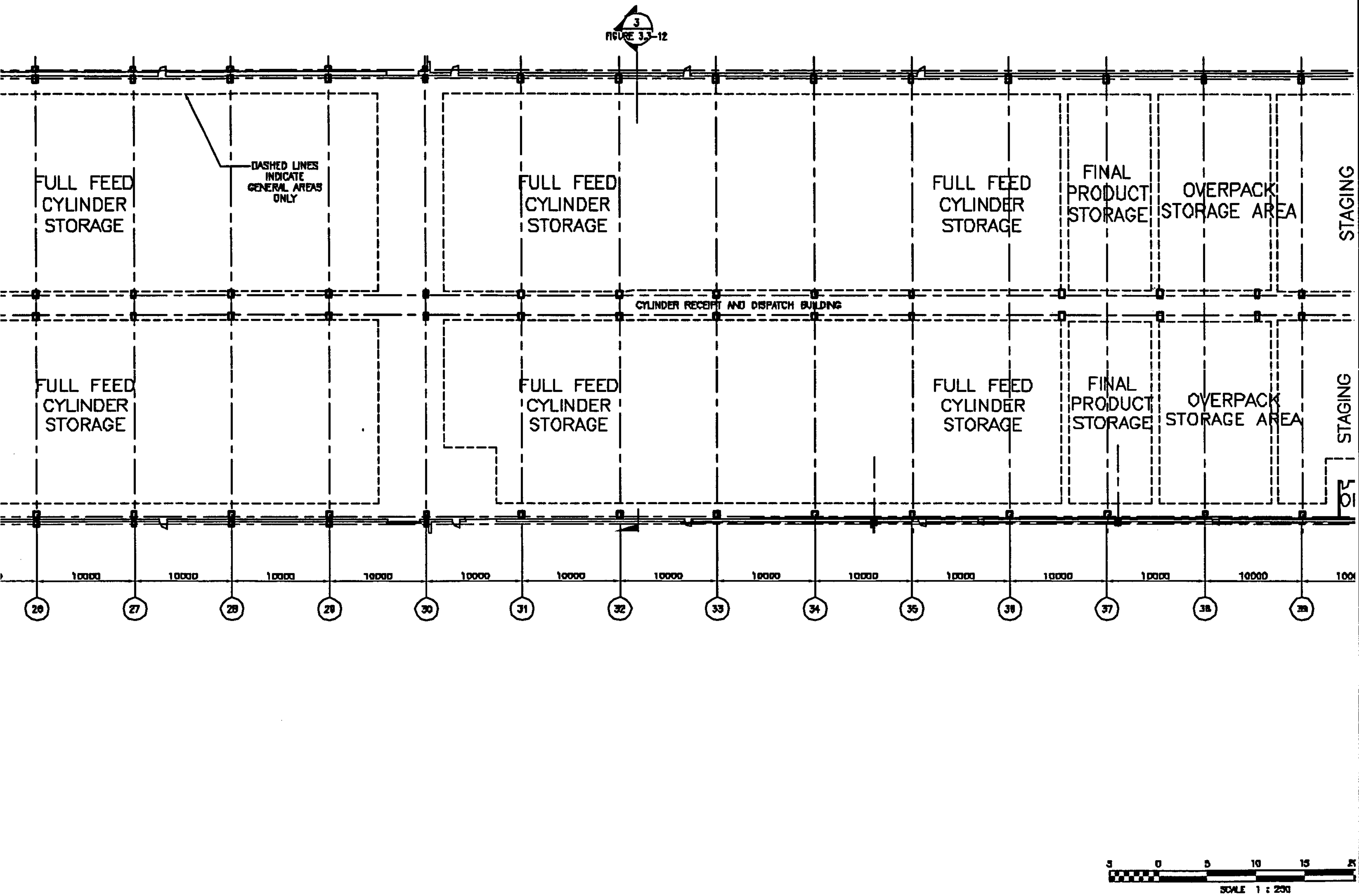
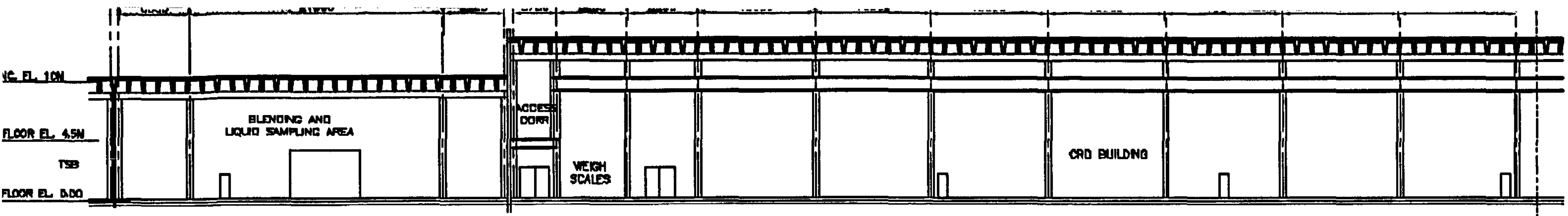
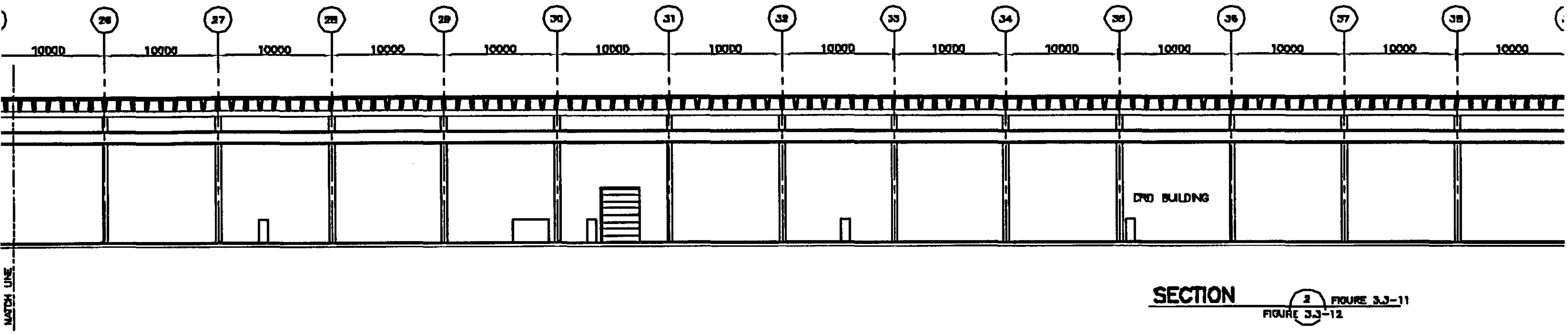
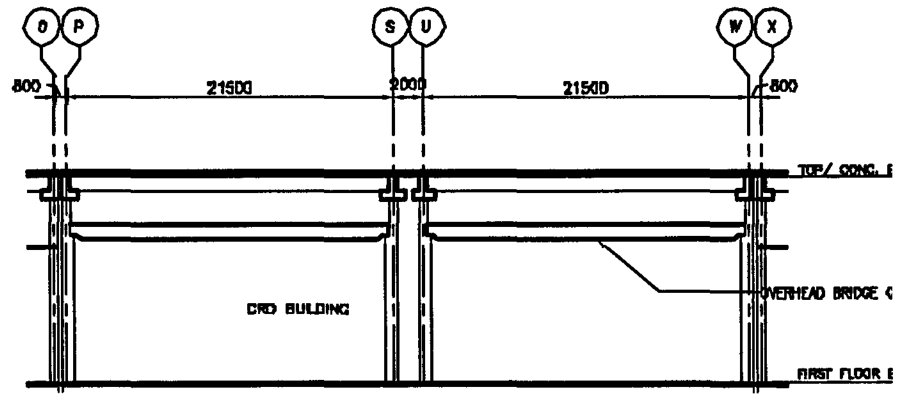

3
FIGURE 3.3-12
DASHED LINES INDICATE GENERAL AREAS ONLY
FULL FEED CYLINDER STORAGE
FULL FEED CYLINDER STORAGE
FULL FEED CYLINDER STORAGE
FINAL PRODUCT STORAGE
OVERPACK STORAGE AREA
STAGING
CYLINDER RECEIPT AND DISPATCH BUILDING
FULL FEED CYLINDER STORAGE
FULL FEED CYLINDER STORAGE
FULL FEED CYLINDER STORAGE
FINAL PRODUCT STORAGE
OVERPACK STORAGE AREA
STAGING
10000
26
27
28
29
30
31
32
33
34
35
36
37
38
39
SCALE 1 : 250



SECTION 1 FIGURE 3.3-17 FIGURE 3.3-10 FIGURE 3.3-12



SECTION 2 FIGURE 3.3-11 FIGURE 3.3-12



SECTION 3 FIGURE 3.3-11 FIGURE 3.3-12

5 0 5 10 15

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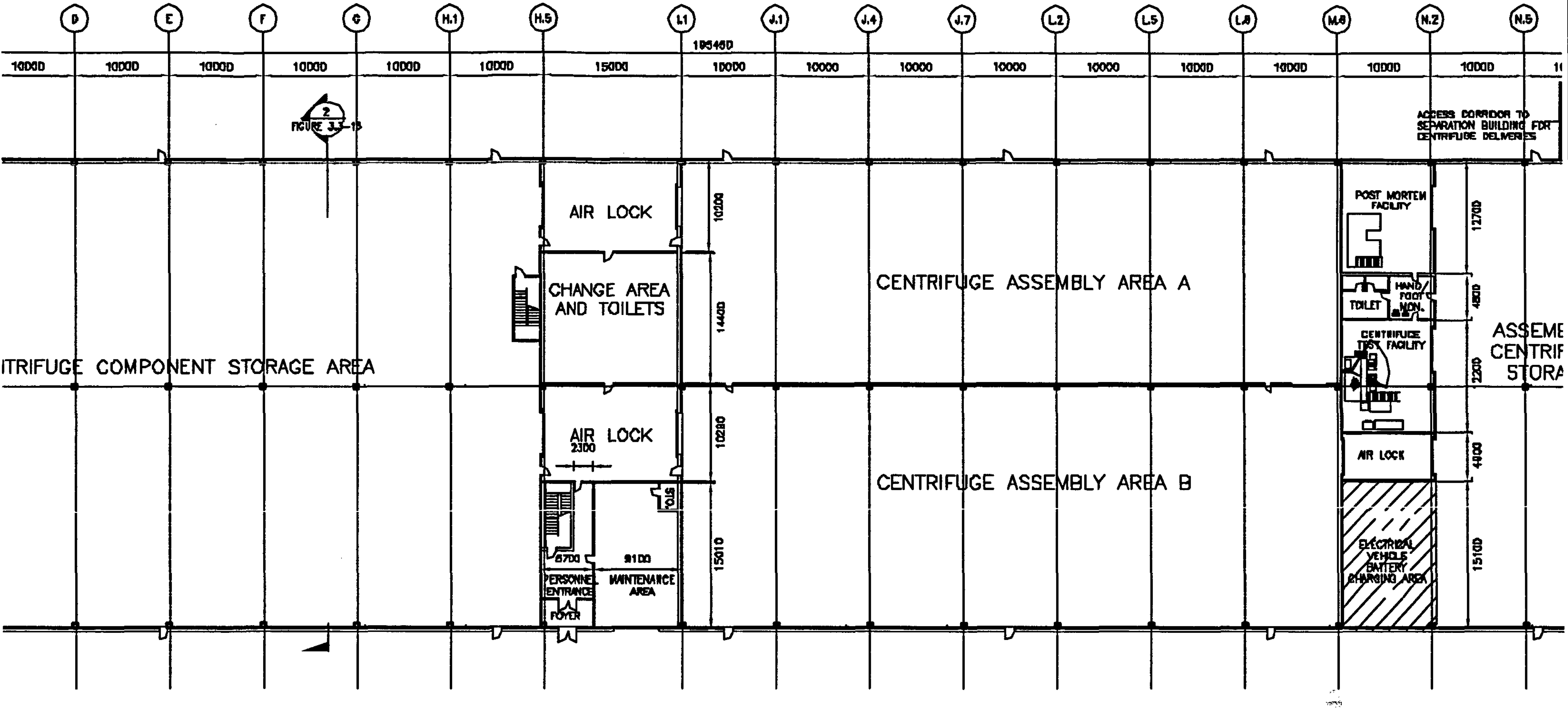

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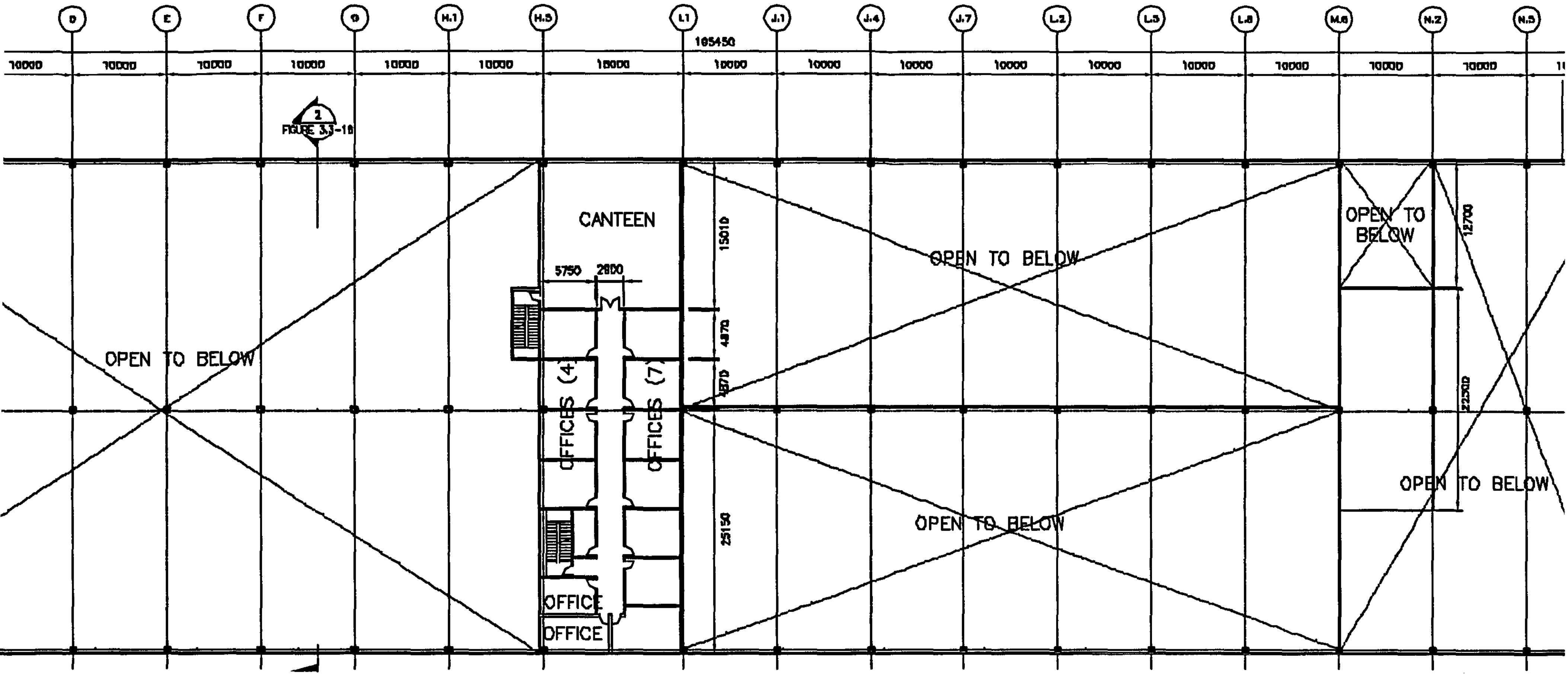
FOR CLARITY



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FOR CLARITY

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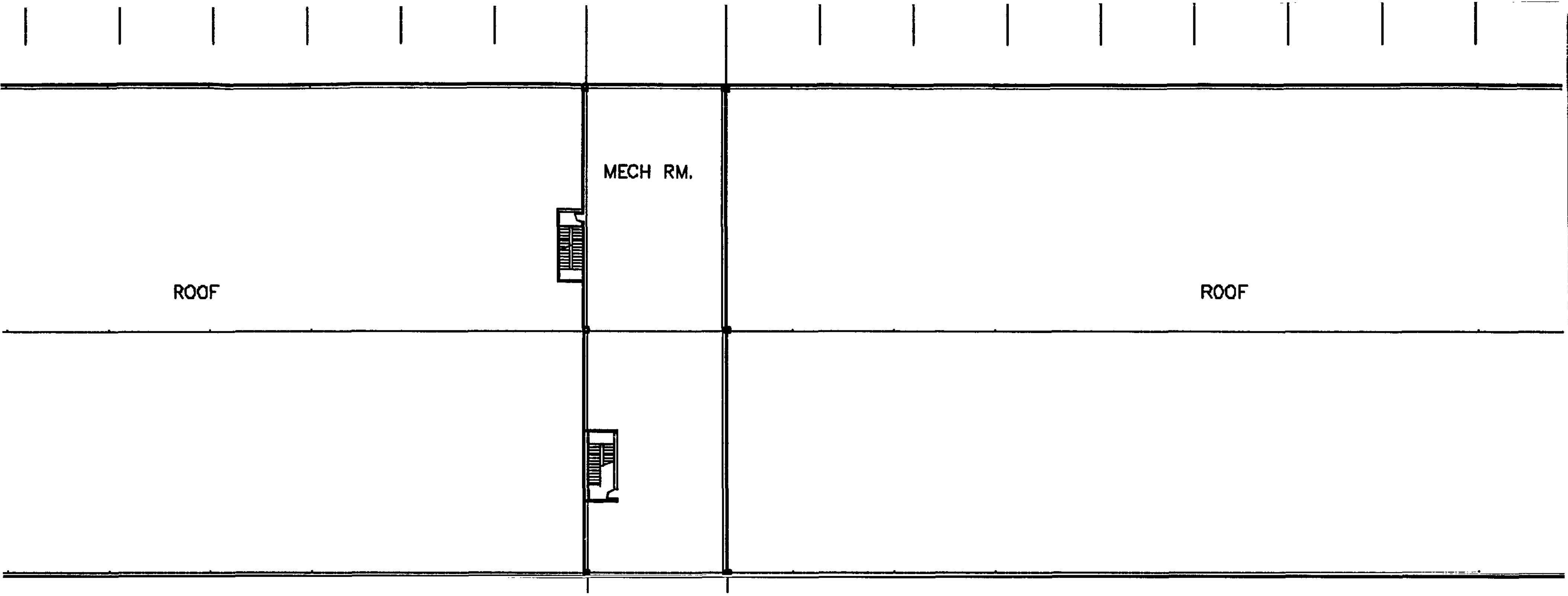

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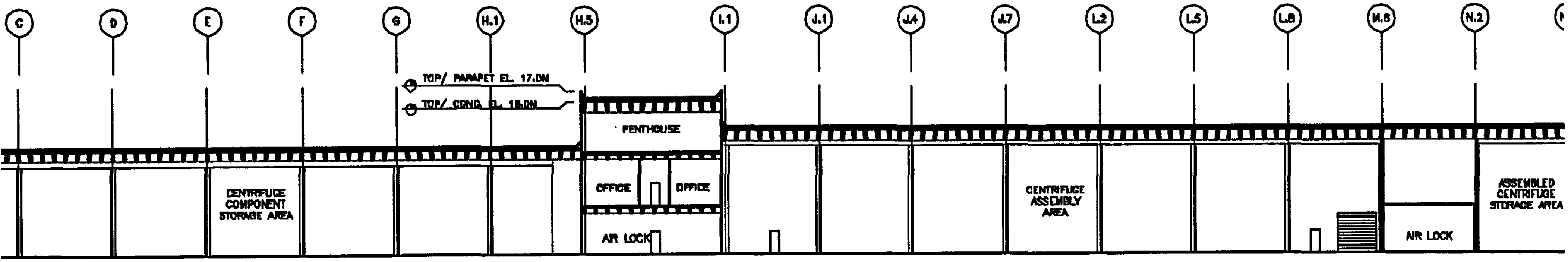
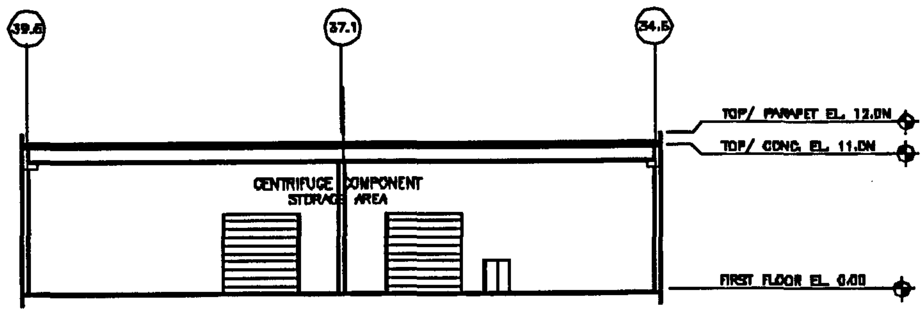
FOR CLARITY



SECTION 1 FIGURE 3.3-16
FIGURE 3.3-13
FIGURE 3.3-14
FIGURE 3.3-13



SECTION 2 FIGURE 3.3-16
FIGURE 3.3-14
FIGURE 3.3-13

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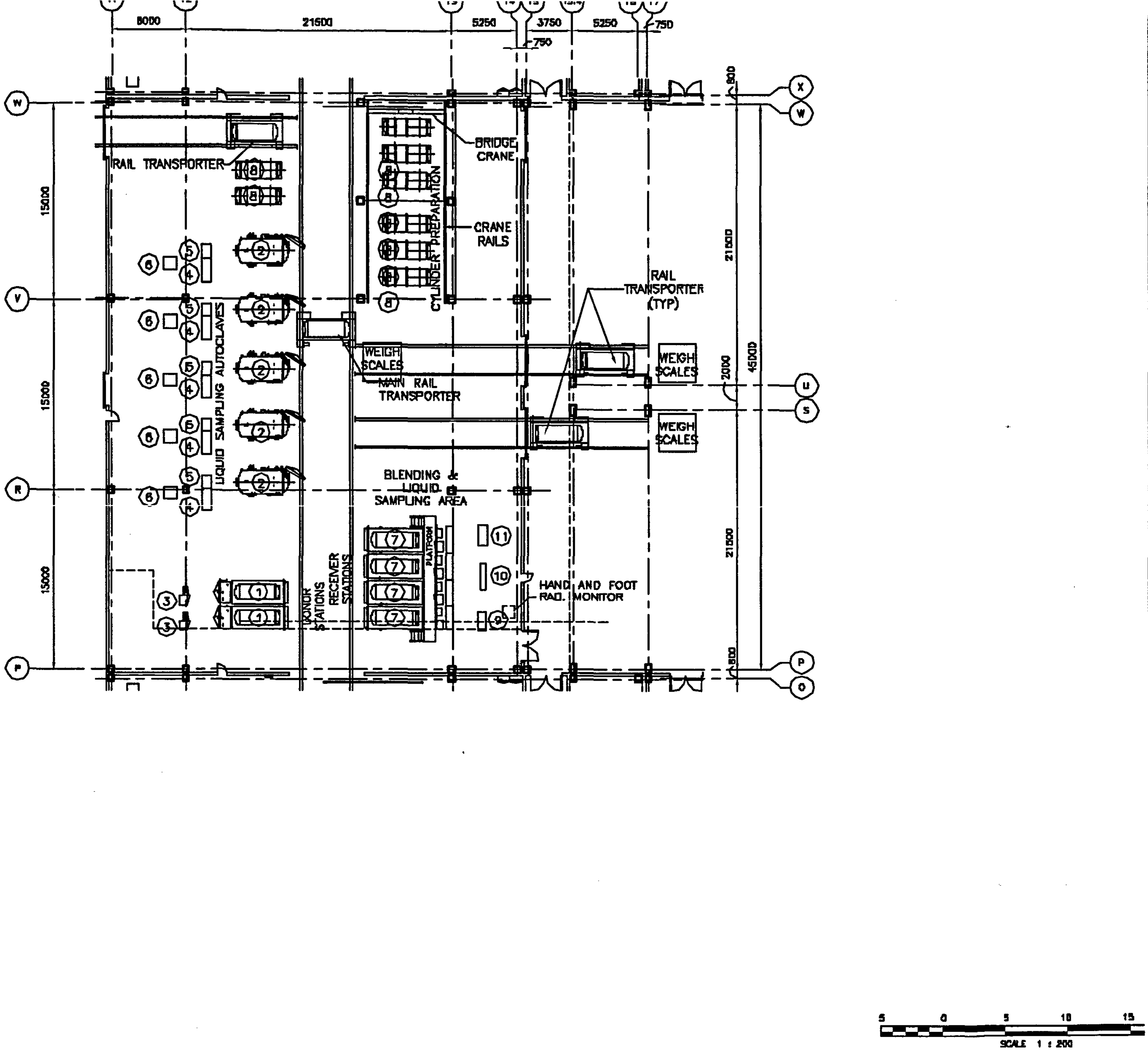

6000
21500
5250
3750
5250
750
750
RAIL TRANSPORTER
BRIDGE CRANE
CRANE RAILS
CYLINDER PREPARATION
LIQUID SAMPLING AUTOCLAVES
RAIL TRANSPORTER (TYP)
WEIGH SCALES
MAIN RAIL TRANSPORTER
WEIGH SCALES
WEIGH SCALES
BLENDING & LIQUID SAMPLING AREA
PLATFORM
DONOR STATIONS
RECEIVER STATIONS
HAND AND FOOT RAD. MONITOR
15000
15000
15000
21500
21500
45000
2000
800
800
W
V
R
P
X
W
U
S
P
O
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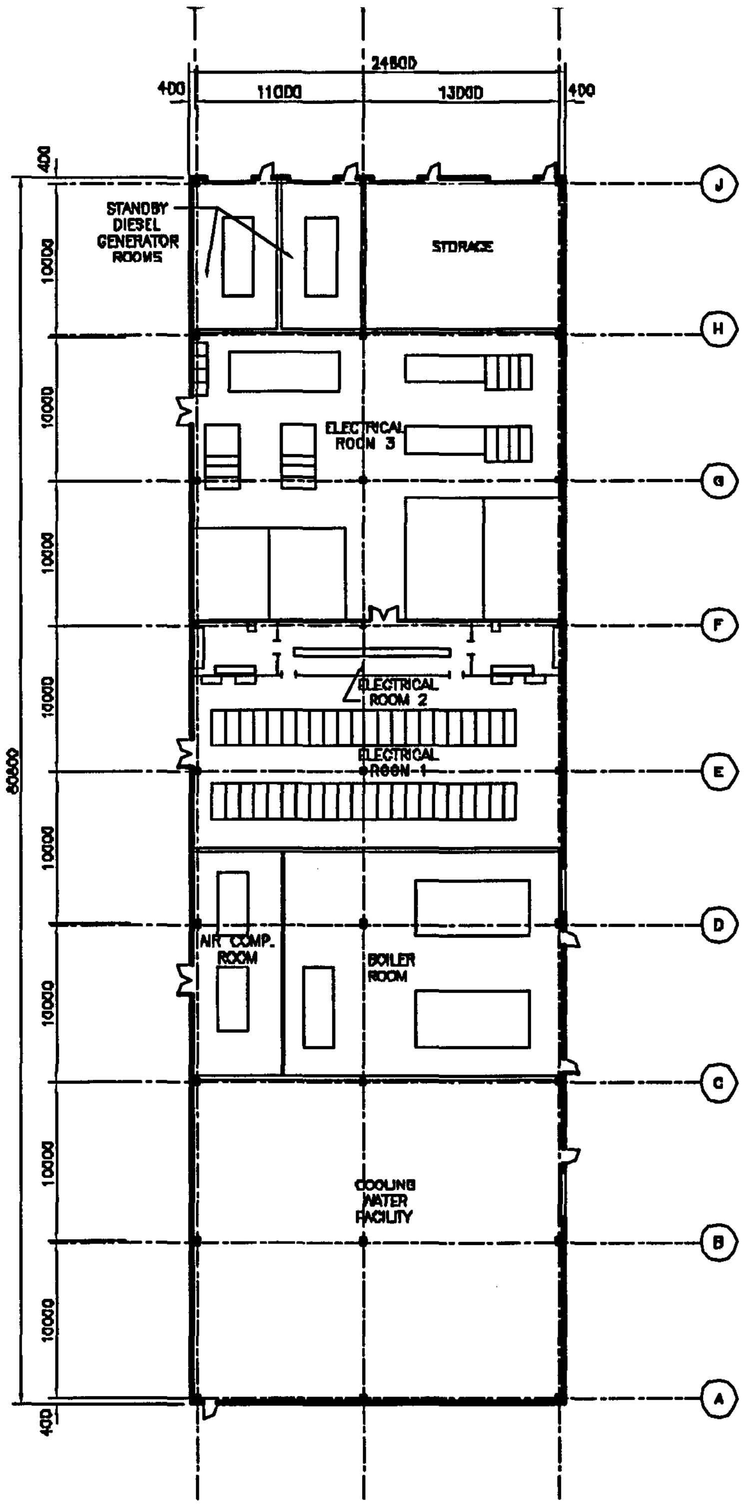

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400
11000
13000
400
400
10000
10000
10000
10000
10000
10000
10000
10000
80800
400
STANDBY DIESEL GENERATOR ROOMS
STORAGE
ELECTRICAL ROOM 3
ELECTRICAL ROOM 2
ELECTRICAL ROOM 1
AIR COMP. ROOM
BOILER ROOM
COOLING WATER FACILITY
J
H
G
F
E
D
C
B
A

5
0
5
10
15
20
SCALE 1 : 250

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3.4 PROCESS DESCRIPTIONS

This section provides a description of the enrichment processes and systems analyzed as part of the integrated safety assessment. A brief overview of the entire enrichment process is provided followed by a detailed description of each process system. The section provides design, operational, and process flow information to support the hazard and accident analysis, as well as to assist in understanding the overall design and function of the National Enrichment Facility (NEF).

The enrichment systems are comprised of the following four major systems:

- UF_6 Feed System
- Cascade System
- Product Take-off System
- Tails Take-off System.

The above systems are used only for the enrichment process. In addition to the four primary systems listed above, there are several major support systems discussed in this section:

- Product Blending System
- Product Liquid Sampling System
- Contingency Dump System.

Finally, the following processes and systems are discussed based on their supporting relationship to the enrichment process and the handling of UF_6:

- Gaseous Effluent Vent Systems (GEVSs)
- Centrifuge Test Facility and Centrifuge Post Mortem Facility
- Material Handling.

Each of the sections that discuss the 10 processes identified above are generally organized to present the following information:

- Functional Description
- Major Components
- Design Description
- Interfaces
- Design and Safety Features
- Operating Limits
- Instrumentation

Items relied on for safety associated with the processes and systems identified above are listed in Section 3.8, Items Relied On For Safety (IROFS).

The calculated values of k_{eff} provided in the following sections were obtained using the criticality code MONK8A (SA, 2001), in conjunction with the JEF2.2 nuclear data library. All values of k_{eff} given in the following sections are equal to $k_{calc} + 3\ \sigma_{calc}$ with a safety limit of 0.95.

In the following sections, the design process parameter values are specified with a datum of standard atmospheric pressure at sea level. These values will be finalized to reflect the site-specific NEF elevation during the design phase and the ISA Summary will be revised accordingly.

The enrichment process at the NEF is basically the same process described in the Safety Analysis Report for the Claiborne Enrichment Center (LES, 1993). The Nuclear Regulatory Commission (NRC) documented its review of the Claiborne Enrichment Center license application and concluded that Louisiana Energy Services' (LES) application provided an adequate basis for safety review of facility operations and that construction and operation of the Claiborne Enrichment Center would not pose an undue risk to public health and safety (NRC, 1994). The design of the NEF incorporates the latest design and safety features from the Urenco enrichment facilities currently operating in Europe.

The major process design differences between the Claiborne Enrichment Center and the NEF are summarized below. Additional details are provided at the beginning of each subsection on how NEF compares to the Claiborne Enrichment Center processes and systems.

The primary difference between the Claiborne Enrichment Center and the NEF is the increase in enrichment capacity. The NEF is designed for 3.0 million separative work units (SWU) per year. The Claiborne Enrichment Center was designed for 1.5 million SWU per year.

The Claiborne Enrichment Center used a feed system that operated above atmospheric pressure. During purification or when feeding to the centrifuges, the UF_6 in the cylinders was in a liquid phase. Autoclaves were used to heat the feed cylinders and to contain any UF_6 in the event there were any leaks in the feed cylinder or piping. The feed purification station used chilled water at 3.9°C (39.0°F) supplied from a common system to chill the purification cylinder.

The NEF feed and feed purification systems do not use UF_6 in the liquid phase. Also, the operating pressure in the feed and purification systems stays considerably below atmospheric pressure. The UF_6 feed is changed from the solid phase to the gaseous phase without going through the liquid phase. This is achieved because the feed system temperature is maintained below the triple point. The Solid Feed Stations used in the UF_6 Feed System are not constructed as autoclaves. There is no need for secondary confinement barriers due to the UF_6 not being in the liquid phase and the subatmospheric pressure of the system. The Feed Purification Low Temperature Take-off Station is cooled by air that is chilled by individual electrically operated chiller units (not water). The purification stations operate at –25°C (-13°F), which is considerably colder than the Claiborne Enrichment Center design. Not using liquid UF_6 and operating at a subatmospheric pressure are major safety enhancements from the Claiborne Enrichment Center design.

The Claiborne Enrichment Center used cooled air at 10°C (50°F) to chill the product cylinders while they were in the Product Take-off Stations. For the NEF, the Product Take-off Stations are cooled by air that is chilled by individual electrically operated chiller units. The Low Temperature Take-off Stations operate at –25°C (-13°F), which is considerably colder than the previous design. The operating pressure for the Low Temperature Take-off Stations is considerably lower for the NEF.

3.4.1 Overview Of Gas Centrifuge Enrichment Process

The function of the NEF is to enrich (increase) the amount of ^{235}U isotope in uranium hexafluoride (UF_6) from naturally occurring feed at 0.711 $^w/_o$ up to a maximum of 5.0 $^w/_o$. The enriched UF_6 is then used for manufacturing fuel for commercial electricity generating nuclear power plants.

Figure 3.4-1, Pictorial Representation of the Enrichment Process, illustrates the process flow in schematic form. An overview of the enrichment process systems and the enrichment support systems are discussed below. Additional details on each of the enrichment process systems are provided in subsequent sections.

3.4.1.1 UF_6 Feed System

The first step in the process is the receipt of the feed cylinders and preparation to feed the UF_6 through the enrichment process.

Natural UF_6 feed is received at the NEF in Department of Transportation (DOT) 7A, Type A cylinders from a conversion plant. The cylinders are ANSI N14.1 (ANSI, applicable version), 48Y or 48X cylinders. Pressure in the feed cylinders is below atmospheric (vacuum) and the UF_6 is in solid form.

The function of the UF_6 Feed System is to provide a continuous supply of gaseous UF_6 from the feed cylinders to the cascades. The maximum feed flow rate is 187 kg/hr (412 lb/hr) based on a maximum capacity of 545,000 SWU/yr per Cascade Hall.

To begin the enrichment process, a 48-in feed cylinder is placed into a Solid Feed Station. There are six Solid Feed Stations per Cascade Hall. Normally three are online. Each Solid Feed Station consists of an insulated enclosure, heated by electric heaters, into which the cylinder is placed. The cylinder is heated to 53°C (127°F) in the Solid Feed Station. At this temperature and pressure (subatmospheric), the solid UF_6 sublimes into a gas. An important safety feature of the feed system is that at no time does the UF_6 go into a liquid phase.

The feed purification system is used to remove the light gas components from the UF_6 feed material to a specified level prior to admittance to the cascades. This protects the centrifuges against high intake of light gas and enhances cascade efficiency by limiting impurities.

For each Cascade Hall, there are two feed purification Low Temperature Take-off Stations. These stations consist of insulated enclosures that are maintained at –25°C (-13°F) by electrically operated chiller units. 48X or 48Y cylinders are placed into the Low Temperature Take-off Station and chilled to –25°C (-13°F). As the gaseous UF_6 enters the cylinder, desublimation into solid UF_6 occurs. In addition to the Low Temperature Take-off Station, there are two UF_6 Cold Traps which desublime UF_6, carbon traps, aluminum oxide (Al_2O_3) traps, and vacuum pumps, used to transfer residual light gas to the Gaseous Effluent Vent System. The carbon and aluminum oxide traps remove trace UF_6 and HF from the gas stream.

After purification, the UF_6 gas is then fed through a main header to the cascades, where the enrichment process actually occurs. The pressure in the main header is limited to 65 mbar (26.1 in. H_2O) to prevent the gaseous UF_6 from desubliming back to a solid at ambient temperature.

3.4.1.2 Cascade System

The function of the Cascade System is to receive gaseous UF_6 from the UF_6 Feed System and enrich the ^{235}U isotope in the UF_6 to a maximum of 5 $^w/_o$.

Multiple gas centrifuges make up arrays called cascades. The cascades separate gaseous UF_6 feed with a natural uranium isotopic concentration into two process flow streams – product and tails. The product stream is the enriched UF_6 stream. The tails stream is UF_6 that has been depleted of ^{235}U isotope.

3.4.1.3 Product Take-off System

The function of the Product Take-off System is to provide continuous withdrawal of the enriched gaseous UF_6 product from the cascades. The maximum product flow rate per Cascade Hall is 18.4 kg/hr (40.6 lb/hr) based on a maximum Cascade Hall capacity of 545,000 SWU/yr.

The product streams leaving the cascades (at each Cascade Hall) are brought together into one common manifold. The product stream is transported via a train of vacuum pumps to Product Low Temperature Take-off Stations. There are five Product Low Temperature Take-off Stations per Cascade Hall. Normally two are on-line when using 30B cylinders. Each Low Temperature Take-off Station consists of an insulated enclosure that is maintained at –25°C (-13°F) by electrically operated chiller units. A 30B or a 48Y cylinder is placed into the Low Temperature Take-off Station and cooled to –25°C (-13°F). The 30B cylinders contain final product to be shipped to the customer. The 48Y cylinders are used internal to the plant for blending purposes. As the enriched gaseous UF_6 enters the cylinder, desublimation into solid UF_6 occurs.

The entire system operates at subatmospheric pressure.

The Product Take-off System also contains a system to purge and dispose of light gas impurities from the enrichment process. This system consists of product vent UF_6 Cold Traps into which UF_6 desublimes while leaving the light gas in a gaseous state. The UF_6 Cold Trap is followed by product vent vacuum pump/chemical trap sets, each consisting of a carbon trap, an aluminum oxide trap, and a vacuum pump. The carbon trap removes small traces of UF_6 and the aluminum oxide trap removes any HF from the gas flow.

There are connections to the Assay Sampling System and the On-line Mass Spectrometer System for product sampling and analysis.

3.4.1.4 Tails Take-off System

The primary function of the Tails Take-off System is to provide continuous withdrawal of the gaseous UF_6 tails from the cascades. The maximum tails flow rate is 168 kg/hr (370 lb/hr) based on a maximum Cascade Hall capacity of 545,000 SWU/yr. A secondary function of this system is to provide a means for removal of UF_6 from the centrifuge cascades under abnormal conditions.

The tails stream exits each cascade via a primary header, goes through a pumping train, and then to Tails Low Temperature Take-off Stations. There are ten Low Temperature Take-off Stations per Cascade Hall. Under normal operation, seven of the Low Temperature Take-off Stations are in operation receiving tails and three are on standby.

Each Low Temperature Take-off Station consists of an insulated enclosure that is maintained at –25°C (-13°F) by electrically operated chiller units. 48Y cylinders are placed into the Low Temperature Take-off Stations and cooled to –25°C (-13°F). As the gaseous depleted UF_6 (tails) enters the cylinder, it desublimes into solid UF_6.

The entire system operates at subatmospheric pressure.

The Tails Take-off System also has an evacuation pump/chemical trap set, and connections to the Assay Sampling Subsystem and an On-line Mass Spectrometer System for continuous gas sampling.

3.4.1.5 Product Blending System

The primary function of the Product Blending System is to provide means to fill 30B cylinders with UF_6 at a specific enrichment of ^{235}U to meet customer requirements. This is accomplished by blending (mixing) UF_6 at two different enrichment levels to one specific enrichment level. The system can also be used to transfer product from a 30B or 48Y cylinder to another 30B cylinder without blending.

The Product Blending System is sized for the complete 3,000,000 SWU/year enrichment plant production.

This system consists of Blending Donor Stations (which are similar to the Solid Feed Stations) and Low Temperature Take-off Blending Receiver Stations (which are similar to the Low Temperature Take-off Stations described earlier).

The donor system consists of two Blending Donor Stations. Each station consists of an insulated enclosure (similar to the Solid Feed Station enclosures). Full 30B or 48Y product cylinders at various enrichment levels are placed into the Blending Donor Stations and are heated to sublime the solid UF_6 to gas. The sublimed gas from the two Blending Donor Stations is transported to four Blending Receiver Stations. Each Blending Receiver Station consists of an insulated enclosure that is maintained at –25°C (-13°F) by electrically operated chiller units. Empty 30B cylinders are placed into the station and cooled to –25°C (-13°F). As the gaseous UF_6 from the Blending Donor Stations enters the cylinder, desublimation into solid UF_6 occurs.

There are no vacuum pumps used to transfer product in this system. The system has a vent system similar to the product vent system.

3.4.1.6 Product Liquid Sampling System

The function of the Product Liquid Sampling System is to obtain a representative assay sample from filled product cylinders. The sample is used to validate the exact enrichment level and quality of UF_6 in the filled product cylinders, before the cylinders are sent to the fuel processor.

This is the only system in the NEF that changes solid UF_6 to liquid UF_6.

The main piece of equipment used in this system is the Product Liquid Sampling Autoclave. A filled 30B product cylinder is placed into the autoclave and a manifold (inside the autoclave) with three sample bottles, is connected to the cylinder valve. After closing the autoclave door, the autoclave is heated to 70°C (158°F) via air heated with electric heaters. As the temperature of the UF_6 in the cylinder increases, the pressure also increases. When the pressure in the sample manifold reaches approximately +2.5 bar (36.3 psia), the temperature is stabilized. At this point, the UF_6 is a liquid. In order to assure that a sample represents the entire contents of the cylinder, it is necessary to homogenize the UF_6. The UF_6 will homogenize when the UF_6 becomes liquid at the high pressure and temperature. Homogenization typically lasts for 16 hours. After the homogenization period, the sampling process is initiated.

After homogenization, with the sample bottle valves closed, the autoclave is tilted via a tilting mechanism to 30 degrees from horizontal. After the sample manifold is filled, the autoclave is lowered to horizontal, and the sample bottle valves are opened and closed in sequence to collect the samples. The autoclave and cylinder is then cooled down and the autoclave is vented and opened for sample bottle removal.

One of the main safety features of the autoclave is that it is designed to provide a secondary confinement barrier in the unlikely event a leak should occur in the UF_6 cylinder or connected piping while the UF_6 is in liquid form. Numerous controls are designed into the autoclave to mitigate overheating and other conditions that may affect the integrity of the UF_6 system.

3.4.2 UF_6 Feed System

The NEF UF_6 Feed System uses a process similar to the original LES Claiborne Enrichment Center. The primary differences are as follows:

A. Feed Station Operating Conditions.

The Claiborne Enrichment Center used a feed station that operated above atmospheric pressure. UF_6 in the feed cylinder was maintained in the liquid phase. Normal UF_6 pressure in the feed cylinder was above atmospheric, at 1.8 bar (26.1 psia). Normal station heating temperature was up to 110°C (230°F). The Claiborne Enrichment Center used a sealed autoclave for secondary containment of the feed cylinder to prevent exposure in the event a leak developed in the primary containment (cylinder and piping).

The NEF sublimes solid UF_6 directly to gaseous UF_6 at subatmospheric pressure, without entering the liquid phase. Normal feed cylinder pressure is 500 mbar (7.25 psia) and the station temperature during heating is limited to 61°C (142°F). As a result, a Solid Feed Station is used to heat the feed cylinder rather than an autoclave.

B. Feed Purification Low Temperature Take-off Cylinder Operating Temperature.

The Claiborne Enrichment Center cylinder temperature was maintained at +3.9°C (39°F) by spraying the cylinder with chilled water. The NEF chills the cylinder to -25°C (-13°F) by using cold air from a refrigeration unit.

3.4.2.1 Functional Description

The principal function of the UF_6 Feed System is to provide a continuous supply of gaseous uranium hexafluoride (UF_6) from the feed cylinders to the cascades. Sublimation from the solid phase, at pressures significantly below atmospheric, is the process used in the UF_6 Feed System. Purification of the as-received UF_6 feed material is accomplished in the Feed Purification Subsystem, where light gas components, primarily air and hydrogen fluoride (HF), are removed. This protects the centrifuges against excessive intake of light gas, which improves cascade production efficiency. Secondary functions of the Feed Purification

Subsystem are to vent the light gas from the system during cylinder changeouts and to remove the final quantity of UF_6 (the heel) from the feed cylinder. The system is shown in Figure 3.4-2, Process Flow Diagram, UF_6 Feed System.

The system produces intermittent gaseous effluent from UF_6 purification operations. Additional small intermittent quantities of gaseous effluent are produced from purging and evacuating the flexible piping used to connect the feed and feed purification cylinders. These effluents are treated by the Feed Purification UF_6 Cold Traps and Vacuum Pump/Chemical Trap Sets to remove UF_6 and HF before being routed to the Separations Building Gaseous Effluent Vent System (GEVS) for further treatment. Solid wastes are produced from periodic change-out of chemical and oil traps. There are no liquid effluents directly produced in this system. Vacuum pumps are taken out of service for maintenance and the pump oil is reprocessed in the Technical Services Building (TSB) and reused.

The UF_6 Feed Systems are located in the UF_6 Handling Area of each Separations Building Module. The location of the major equipment is shown on Figure 3.3-2, Separations Building Module, First Floor and Figure 3.3-3, UF_6 Handling Area, Equipment Location. The UF_6 Feed Systems are operated from the Control Room, with the exception of maintenance and preparation activities, which are controlled locally.

3.4.2.2 Major Components

The major components of the UF_6 Feed System are described below.

A. Solid Feed Station.

A Solid Feed Station consists of an insulated box with a non-flammable core, complete with rails for the electric carriage of the cylinder transporter. A Solid Feed Station is shown in Figure 3.4-3, Solid Feed Station Equipment Drawing. Each Solid Feed Station incorporates an electric air heater and circulation fan, with controls, to provide thermal energy to the solid UF_6 to cause it to sublime within the cylinder. A weighing device is provided in the Solid Feed Station (a frame with four load cells) to provide continuous on-line weighing of UF_6 in the feed cylinder.

The front of the Solid Feed Station is made up of a single door. Connection of the cylinder in a Solid Feed Station is made at the front (door) end. The Solid Feed Station does not have an opening at the back. Rubber seals are used on the openings in the Solid Feed Station to minimize leaks for energy conservation.

B. Solid Feed Station Valve Hotbox.

Valves in a Solid Feed Station Valve Hotbox connect the feed cylinder to the Main Feed Header, the Feed Purification Subsystem, or the Nitrogen System. Manual and automatic isolation valves, a pressure control valve, and pressure transducers are contained in the electrically heated hotboxes to maintain them at a stable temperature. The UF_6 piping between the Solid Feed Station and hotbox is heat traced.

C. Main Feed Header.

The Main Feed Header connects the Solid Feed Station Valve Hotboxes to each of the cascades in a Cascade Hall. Pressure is controlled in the header so that heat tracing is not required.

D. Feed Purification Subsystem.

A Feed Purification Subsystem is provided for each Cascade Hall and consists of two Low Temperature Take-off Stations, each with associated valve hotbox, UF_6 cold trap and heater chiller unit, and a vacuum pump/chemical trap set. One Feed Purification Subsystem is provided for each Cascade Hall, but each major component in the system is duplicated. The major components of the Feed Purification Subsystem are described below:

1. Low Temperature Take-off Station (LTTS). A LTTS consists of a composite panel box construction complete with rails for the electric carriage of the cylinder transporter. An LTTS is shown in Figure 3.4-4, Low Temperature Take-off Station Equipment Drawing. The box panels have a non-flammable insulated core and are vapor sealed to prevent ice build-up within the insulation. Each LTTS incorporates an air chiller unit, with controls, to remove thermal energy from the UF_6 gas to cause it to desublime in the cylinder. The chiller unit has a defrost cycle, using a heater, to prevent ice buildup on the coils. A hot air blower directed at the cylinder valve prevents UF_6 from desubliming and blocking the cylinder inlet. A weighing device is provided in the LTTS (a frame with four load cells and associated instrumentation) to provide continuous on-line weighing of UF_6 in the purification cylinder.

   The front of the LTTS is made up of a single door and the back is furnished with an opening to facilitate connection of the cylinder to the UF_6 piping. A rubber bellows is fitted around the back opening, which envelops the cylinder valve, to prevent cooled air from leaking out of the LTTS. Similar seals on the other openings in the LTTS minimize leaks for energy conservation. The LTTS access openings are provided with heat tracing to prevent ice build-up.

2. Low Temperature Take-off Station Valve Hotbox. Valves in a hotbox connect the LTTS to the Solid Feed Station Valve Hotboxes, the UF_6 cold traps, or the Nitrogen System. Manual and automatic isolation valves and a pressure transducer are contained in the electrically heated hotboxes to maintain them at a stable temperature. The UF_6 piping between the Solid Feed Station Valve Hotboxes and the LTTS Valve Hotboxes is heat traced.

3. UF_6 Cold Trap. Each UF_6 cold trap consists of an insulated horizontal tube with internal baffles. A UF_6 cold trap is shown in Figure 3.4-5, UF_6 Cold Trap Equipment Drawing. The UF_6 cold trap has a dedicated heater/chiller unit operating at a cooling set point and a heating set point. Each heater/chiller unit contains approximately 70 L (19 gal) of silicon oil, as the heat exchange media, which circulates around each cold trap. These Feed Purification Subsystem heater/chiller units are separated by over 30 m (100 ft) from other heat/chiller units in similar subsystems. The low temperature removes the thermal energy from the UF_6 gas, causing it to desublime on the internal walls of the trap, while leaving the light gas in the gaseous phase. The high temperature results in sublimation of the UF_6 contents of the UF_6 cold trap for transfer back to a feed purification cylinder. Each end of the UF_6 cold trap is heat traced to prevent the UF_6 from solidifying and blocking the UF_6 cold trap entrance or exit. The UF_6 cold trap has a weighing device to provide continuous on-line weighing of the UF_6 accumulated.

   An automatic control valve located after each UF_6 cold trap restricts the flow of gases through the UF_6 cold traps. This ensures an adequate residence time for the gases in the UF_6 cold trap to allow all of the UF_6 to desublime.

4. Vacuum Pump/Chemical Trap Set. The UF_6 cold traps are followed by vacuum pump/chemical trap sets. Each set has a carbon trap, an aluminum oxide trap, an insulated vacuum pump with nitrogen purge, and an oil trap on either side of the vacuum pump. A chemical trap is shown in Figure 3.4-6, Chemical Trap Equipment Drawing. The vacuum pump exhausts into the Separations Building GEVS. The activated carbon trap removes small traces of UF_6. The aluminum oxide trap removes HF. Oil traps are installed before and after the vacuum pump to prevent oil migration both upstream and into the Separations Building GEVS.

3.4.2.3 Design Description

The design bases and specifications are given in Table 3.4-1, UF_6 Feed System Design Basis. Applicable Codes and Standards are given in Table 3.4-2, UF_6 Feed System Codes and Standards.

Each UF_6 Feed System is dedicated to an individual Cascade Hall of eight cascades. Gaseous UF_6 feed (natural, 0.711 $^{w}/_{o}$ ^{235}U) flows from the Solid Feed Stations to the centrifuge cascades. The system is designed to provide a total maximum Cascade Hall flow rate of 187 kg/h (412 lb/hr) based on a capacity of 545,000 SWU/ year. A single cascade in operation generates a minimum flow rate of 13.5 kg/h (29.75 lb/hr). The peak flow rate for an individual cascade during the feed inlet sequence is 27 kg/h (59.5 lb/hr).

The entire UF_6 Feed System operates at subatmospheric pressure. In the event of a confinement barrier failure (e.g., pipe leak), releases of uranyl fluoride (UO_2F_2) and HF are greatly minimized because air will migrate into the system rather than UF_6 escaping from the system. This important safety feature greatly limits the likelihood of exposures.

There are six Solid Feed Stations, each with an associated valve hot box, connected in parallel to the main feed header in each UF_6 Feed System. At any time three Solid Feed Stations can be on-line to handle the maximum UF_6 feed flow to one Cascade Hall. Two Solid Feed Stations are in either standby mode or preparation mode. The sixth Solid Feed Station is a spare and can be in either standby, off-line, preparation, or maintenance mode.

Each UF_6 Feed System has a dedicated Feed Purification Subsystem, consisting of two LTTSs, two UF_6 Cold Traps, and two Vacuum Pump/Chemical Trap Sets connected in parallel. One of the LTTSs, UF_6 Cold Traps, and Vacuum Pump/Chemical Trap Sets is available for use, while the second is a spare and can be in, off-line, preparation (cylinder being installed or removed), or maintenance mode.

Prior to feeding UF_6 to the cascades, the contents of each cylinder are purified and verified as natural UF_6. This verification is accomplished using distinguishing markings/identification of 48X and 48Y cylinders within the UF_6 area to ensure cylinders containing product are not placed on-line to the cascade and by sampling and assay analysis of a feed cylinder contents for uranic enrichment. Any light gases, primarily air and HF, and a specified quantity of UF_6 are transferred to a purification cylinder, to ensure that impurities are removed from the feed cylinder. Likewise, the purification cylinder is relieved through the UF_6 Cold Trap and Vacuum Pump/Chemical Trap Set to the Separations Building GEVS. Finally a sample of the gaseous UF_6 is desublimed into a sample bottle for analysis.

The Solid Feed Station provides controlled heat to the feed cylinder to sublime the UF_6 directly from solid phase to gaseous phase at subatmospheric pressures. Pressure is controlled throughout the system to maintain the subatmospheric pressures and to provide the required flow rate. UF_6 piping and valve stations where UF_6 desublimation could occur are heated. The building heating and ventilation system is designed to maintain a minimum temperature of 18°C (64.4°F), therefore heat tracing of the main feed header, which is controlled at a pressure less than 65 mbar (26.1 in. H_2O), is not required.

All components and piping in the UF_6 Feed System operate at subatmospheric pressure. Release of UF_6 and/or HF is unlikely because leakage, if it were to occur, would be into the system.

The materials of construction and fabrication specifications for the equipment and piping used in the UF_6 Feed System are compatible with UF_6 at the operating conditions and have been proven by over 30 years of use in existing Urenco European enrichment plants.

3.4.2.4 Interfaces

The UF_6 Feed System interfaces with the following systems and utilities:

A. Cascade System

B. GEVS

C. Nitrogen System

D. Compressed Air System

E. Electrical System

F. Plant Control System

G. Hoisting and Transportation Equipment.

3.4.2.5 Design and Safety Features

The UF_6 Feed System is designed and constructed to provide safe operation for plant personnel as well as the general public. Principal design features are as follows:

A. All process piping, valves, vessels and pumps in the UF_6 Feed System operate at subatmospheric pressure.

B. Piping is all welded construction and process valves are bellows sealed.

C. Before disconnecting any equipment, the process piping is evacuated and purged with nitrogen.

D. A local exhaust to the Separations Building GEVS is provided any time a UF_6 line is disconnected.

E. Before discharge to the Separations Building GEVS, all gases flow across activated carbon and aluminum oxide in the Feed Purification Subsystem vacuum pump/chemical trap set to remove any traces of UF_6 and HF.

F. Temperature in each Solid Feed Station and LTTS is monitored and controlled.

G. Feed purification cylinder overfill is prevented by two weight trips. The first is at the desired net weight of UF_6 and the second is at the gross weight of the cylinder with UF_6 contents. Only the first trip is operator adjustable.

H. Hydrocarbon lubricants are not used. The Feed Purification vacuum pumps are lubricated with fully fluorinated synthetic oil such as "Fomblin," a perfluorinated polyether (PFPE).

I. Removal of a connected cylinder from an LTTS is prevented by an interlock system. Unless the flexible hose on the cylinder valve has been removed and locked in its "holster," a physical barrier prevents the cylinder transporter drawbridge from docking with the station rails, preventing cylinder removal.

J. Temperature in the Feed Purification Subsystem carbon trap is monitored and controlled.

K. Should a blockage occur in a section of process piping, the heat tracing on that section of pipe is not allowed to be switched on until the solid UF_6 has been removed.

L. Mechanical interlocking systems are provided in all solid feed and low temperature stations to prevent the operation of the stations with an incorrect cylinder type loaded. The system prevents the use of cylinders identified for product take-off from being used in either a solid feed station or feed purification station.

3.4.2.6 Operating Limits

The UF_6 Feed System must provide purified feed to the cascades at the minimum and maximum rates under normal operating conditions. A Cascade Hall's normal maximum capacity is based on 545,000 SWU/yr.

3.4.2.7 Instrumentation

The process variables, such as pressure, temperature, and valve positions, are automatically controlled. Deviations from specified values are detected and indicated via a two level alarm system. At the first alarm level, the process operator has the ability to manipulate the process to restore it to normal. At the second alarm level, automatic action is taken to provide system protection. For safety, system protection, and operability, some sensors are duplicated and others are installed in triplicate. Action is initiated if any one out of two or three sensors reach alarm levels.

A. Solid Feed Station

Both the Solid Feed Station air temperature and cylinder temperature are monitored to prevent over pressurization of the feed cylinder due to overheating. Normal air temperature in the Solid Feed Station during heating ranges from ambient to 61°C (142°F), while the cylinder temperature ranges from ambient to 53°C (127°F). The first alarm level is 62°C (144°F) for the Solid Feed Station air and 54°C (129°F) for the cylinder to give the operator warning of high temperature. The second alarm level is 55°C (131°F) for the cylinder, which trips the Solid Feed Station heater off.

In addition to the temperature controls, the Solid Feed Station has two independent and diverse temperature protection instruments. One is failsafe hard wired and measures cylinder temperature, and the other is a failsafe capillary type and measures the Solid Feed Station air temperature. These provide extra safety margin to prevent overheating the cylinder if the air temperature control fails. Both systems automatically de-energize the air heater and blower, if either the cylinder temperature reaches 55°C (131°F) or the Solid Feed Station air temperature reaches 63°C (145°F).

The feed cylinder pressure is monitored with dual sensors to prevent over pressurization of the cylinder, piping and valves. Normal pressure is 500 mbar (7.25 psia). The first alarm level is 600 mbar (8.7 psia) to give the operator warning of over pressure. The second alarm level at 850 mbar (12.3 psia) automatically closes the cylinder valve and trips the Solid Feed Station off-line, which de-energizes the air heaters and blower.

Each Solid Feed Station has a weighing system to monitor the contents of the feed cylinder. The first weight trip of 800 kg (1,764 lb) gross is used to indicate a cylinder is present in the Solid Feed Station. The second weight trip, equal to a net UF_6 weight of 100 kg (221 lb), indicates the cylinder is empty and puts the Solid Feed Station in standby.

B. Solid Feed Station Valve Hotbox

A single pressure transducer is located in the piping in each Solid Feed Station Valve Hotbox. When selected to control the Solid Feed Station, it is used to modulate the Solid Feed Station feed control valve. Normal pressure is approximately 55 mbar (22.1 in. H_2O). A first alarm, at 58 mbar (23.3 in. H_2O), warns the operator of high pressure. The second alarm level, at 64 mbar (25.7 in. H_2O), automatically switches the Solid Feed Station to standby and closes the outlet valve.

Low feed pressure is also alarmed. The first alarm, at 50 mbar (20.1 in. H_2O), warns the operator of loss of feed supply. A second alarm at 30 mbar (12.0 in. H_2O) indicates that the feed cylinder is empty.

C. Main Feed Header

Two pressure transducers are located in the main feed header near the Solid Feed Stations. When selected to control a Solid Feed Station, one of the instruments is used to modulate the Solid Feed Station feed control valve. Normal pressure is 55 mbar (22.1 in. H_2O). A first alarm at 57 mbar (22.9 in. H_2O) warns the operator of high pressure. The second alarm level, at 67 mbar (26.9 in. H_2O), automatically switches all of the Solid Feed Stations to standby and closes each Solid Feed Station's outlet valve. A low alarm at 20 mbar (8.03 in. H_2O) warns the operator of loss of feed supply.

In addition, three pressure transducers are evenly distributed along the feed header near the cascades. These act on a one out of three basis to protect the cascades from abnormal pressures. A first high alarm at 57 mbar (22.9 in. H_2O) warns the operator of high pressure. The second high alarm level, at 70 mbar (28.1 in. H_2O), automatically prevents feeding into the cascades. A first low alarm at 50 mbar (20.1 in. H_2O) warns of loss of the feed supply. The second low alarm level, at 20 mbar (8.03 in. H_2O), automatically prevents feeding into the cascades.

D. Feed Purification Low Temperature Take-off Stations

The purification cylinder inlet pressure is monitored to assure that a cylinder is connected to the system. Normal pressure is approximately 50 mbar (20.1 in. H_2O). A first alarm warns of high pressure at 400 mbar (5.8 psia). At 450 mbar (6.53 psia) the LTTS Valve Hotbox inlet valve is closed and the LTTS is tripped to standby. At a pressure below 40 mbar (16.1 in. H_2O) the cylinder is available for feed purification, and below 10 mbar (4.01 in. H_2O) it is available for feed cylinder heel removal.

Each LTTS has a weighing system to monitor the contents of the purification cylinder. The first alarm is 8,500 kg (18,743 lb) net weight for a 48Y type cylinder, above which efficiency is reduced. At 12,400 kg (27,342 lb), the maximum operational net weight for a 48Y type cylinder, the LTTS trips to standby and the inlet valve closes. A second trip at 15,300 kg (33,737 lb) gross weight for a 48Y type cylinder also closes the inlet valve and trips the LTTS off-line. A low alarm at 800 kg (1,764 lb) gross weight indicates no cylinder present in the LTTS. Similar trips and alarms are established for a 48X type cylinder. The output of the weighing system also allows cylinder weight to be verified to be within specified trending limits.

For temperature control and protection from high temperatures, the LTTS has a stand-alone control and protection system. The total system consists of three sensors. For main LTTS temperature control, one sensor is mounted in the air return to the chiller unit and monitors the circulating air temperature. This sensor and local control maintains the LTTS temperature to a normal value of –25°C (-13°F). In addition to controlling the LTTS temperature, one output is monitored by the Plant Control System (PCS) and warns when the air temperature rises from –25°C (-13°F) to –5°C (23°F). This would indicate a chiller failure or that the defrost heater is not functioning properly. The LTTS refrigeration unit has a defrost cycle to remove ice from the cooling coils. This is done with a defrost heater at the coils. When the defrost heater is on, the circulating air fan is off to minimize the increase in LTTS air temperature.

In addition to the closed loop control system previously described, there are two independent and diverse temperature protection instruments. These provide extra safety margin to protect against increases in temperature that may occur if the defrost heater control does not operate properly. The first instrument measures the circulating air temperature and is fail-safe hardwired. The second measures the air inside the LTTS and is a fail-safe capillary device. Both of these instruments will trip the defrost heater and fan power supply in the event the air temperature rises above set points. Set point on the hardwired instrument is 50°C (122°F) and set point on the capillary instrument is 63°C (145°F). If heater trip occurs from these two instruments, the LTTS is automatically taken off-line and put into a standby mode.

To prevent desublimation in the cylinder valve, hot air is blown over the valve with a hot air blower. A temperature sensor on the valve controls the temperature to 63°C (145°F).

E. Feed Purification UF_6 Cold Traps

Dual pressure instruments monitor the UF_6 cold trap inlet pressure. The instruments have different ranges and each is used during different purification operations.

During the purification operation, the UF_6 cold trap outlet pressure is monitored. A first high alarm, at 70 mbar (28.1 in. H_2O), warns of high pressure in the UF_6 cold trap. A first low alarm, at 20 mbar (8.03 in. H_2O), warns of low pressure and indicates the UF_6 cold trap is empty when collected UF_6 is being sublimed for transfer back to a purification cylinder. A second low alarm, at 1 mbar (0.2 in. H_2O), closes the UF_6 cold trap outlet valve to prevent UF_6 flow to the vacuum pump. A second high alarm, at 80 mbar (32.1 in. H_2O), trips the UF_6 cold trap off-line, switching the heater/chiller unit off and closing the inlet and outlet valves.

A pressure sensor and control valve between each UF_6 cold trap and its vacuum pump/chemical trap set restricts the flow of light gases through the UF_6 cold trap to ensure all UF_6 desublimes and does not reach the carbon trap. The line pressure into the vacuum pump/chemical trap set is controlled at 3 mbar (1.2 in. H_2O).

A weighing system monitors the contents of the UF_6 cold trap. A first alarm at 40 kg (88.2 lb) warns that the UF_6 cold trap is approaching capacity. At 50 kg (110 lb) the UF_6 cold trap inlet and outlet valves are closed.

The temperature of the UF_6 cold trap is controlled at –60°C (-76°F) during cooling and at 20°C (68°F) for heating during sublimation to empty the UF_6 cold trap of collected UF_6 (gasback). A low alarm at –63°C (-81°F) warns of a chiller unit fault. A first high alarm at –52°C (-62°F) closes the UF_6 cold trap outlet valve and a second high alarm at 25°C (77°F) warns of high temperature during gasback. At 30°C (86°F) the unit trips off-line to avoid desublimation of UF_6 in the header.

F. Feed Purification Vacuum Pump/Chemical Trap Sets

To prevent the carbon trap from overheating and overfilling with UF_6, there are two instruments. One sensor monitors the carbon trap temperature. This sensor will close the Feed Purification UF_6 cold trap outlet valves when carbon trap temperature exceeds 42°C (108°F). This blocks flow to the vacuum pump/chemical trap set. The carbon trap also has a weigh system. In addition to local weight display, this system will shut down the vacuum pump when the high weight set point is reached. The carbon trap weigh system has an alarm at 6 kg (13.2 lb) to warn the operators the carbon trap is approaching full. The vacuum pump trip occurs at 12 kg (26.5 lb).

The activated aluminum oxide (Al_2O_3) trap on the vacuum pump/chemical trap set is also equipped with a weigh system. The weigh system on the aluminum oxide trap only displays a weight locally. There is no control function on this weigh indicator.

Increase in weight is used to monitor accumulation of UF_6 in the carbon trap and HF in the aluminum oxide trap. The chemical traps are replaced based on the accumulated weight.

3.4.3 Cascade System

The primary difference between the Louisiana Energy Services, Claiborne Enrichment Center, and the NEF is the increase from seven to eight cascades per Cascade Hall. The Cascade System used in the NEF is virtually the same as the Claiborne Enrichment Center Cascade System. The NRC staff previously reviewed the Claiborne Enrichment Center SAR license application relative to the Cascade System and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion on the Cascade System is provided in NUREG-1491 (NRC, 1994), Section 3.4.

3.4.3.1 Functional Description

The function of the Cascade System is to receive gaseous UF_6, with a natural uranium isotopic concentration, from the UF_6 Feed System and separate it into two streams, increasing the ^{235}U isotope content in one, the "product," and decreasing the ^{235}U content in the other, the "tails." These UF_6 streams flow from arrays of gas centrifuges, called cascades, through headers to the Product Take-off System and Tails Take-off System. The enrichment process is illustrated in Figure 3.4-7, The Enrichment Process, and Figure 3.4-8, Cascade Process Scheme Equipment Drawing.

3.4.3.2 Major Components

The major components of the Cascade System are:

A. Centrifuges

The latest qualified centrifuge, Model TC-12, contains a rotor that is used to produce the centrifugal force needed for isotope separation. An electromagnetic motor drives the rotor. A stationary center post in the rotor provides for the input of UF_6 feed and output of UF_6 product and tails. The rotor assembly is inside an aluminum outer casing that is under vacuum. The casing provides a vacuum enclosure outside the rotor to reduce drag. A gas centrifuge is shown in Figure 3.4-9, Principle of a Gas Centrifuge.

B. Centrifuge Drive System

The medium frequency supply system provides the electrical power at the required frequency for the centrifuge drive motors. The system consists of run and run-up solid-state frequency converters, a medium frequency distribution system and 60 Hz electrical supply transformers. The Electrical System is described in Section 3.5.2, Electrical System.

C. Cascade Pipe-work

The arrays of centrifuges that make up a cascade are grouped into blocks; the cascade pipe-work connects these blocks and provide for feed, tails, and product flows.

D. Centrifuge Valve Station

The cascades are connected to the UF_6 Feed System, the Product Take-off System, the Tails Take-off System, and the Contingency Dump System. The associated cascade valves and instrumentation are supported on a cascade dedicated valve station. The valve station also provides connection points for the mobile sampling rig and mobile evacuation rigs.

E. Centrifuge Cooling Water Distribution System

The cascade temperature is controlled by a closed loop cooling water system. The cooling water flows through jacketed coils located at the top and bottom of the outer casing. The cascades are housed within enclosures to maintain optimum temperature conditions. The Centrifuge Cooling Water Distribution System is described in Section 3.5.5.2, Centrifuge Cooling Water Distribution System.

F. Mobile Evacuation Rigs and Sampling Rig

Two Mobile Evacuation Rigs are used to sustain a low pressure in the cascade prior to and during centrifuge run-up or run-down. A Mobile Sample Rig is provided to periodically collect

UF_6 samples from a cascade. The rigs connect to a cascade at the cascade valve station. A rig consists of a liquid nitrogen dewar, a roots vacuum pump, an activated carbon trap, and a rotary vane vacuum pump preceded by an aluminum oxide trap and followed by an oil trap. The sample rig has, in addition, product and tails sample bottles. Rig exhausts are connected to the Separations Building GEVS.

3.4.3.3 Design Description

Arrays of gas centrifuges, called cascades, separate gaseous UF_6 feed, with a natural uranium isotopic concentration, into a product stream enriched in the ^{235}U isotope and a tails stream depleted in the ^{235}U isotope.

Should the UF_6 in a cascade need to be rapidly removed to protect the equipment from a process upset or failure, it is automatically accomplished via the Tails Take-off System. Should this system be unavailable at the time, a Contingency Dump System functions as a backup. A centrifuge monitoring system detects rotor failures, i.e., "crashes," and signals the Control Room.

Each centrifuge has an outer casing which functions as a vacuum chamber to reduce friction on the centrifuge rotor, and acts as a barrier for flying parts should a centrifuge fail.

Mobile evacuation rigs are used to evacuate the cascade prior to startup, for maintenance, and shutdown purposes. A mobile cascade sample rig is provided to periodically collect UF_6 samples from a cascade. The carbon trap of the mobile cascade sample rig has a weighing system that will automatically trip the associated vacuum pump on high carbon trap weight. These rigs are connected at the cascade valve station.

The design bases, codes and specifications used by Urenco in the centrifuge and cascade design provide a large safety margin between normal and accident conditions so that no failures could result in any release of hazardous material. Applicable codes and standards are given in Table 3.4-3, Cascade System Codes and Standards. Operation of hundreds of thousands of centrifuges over many years in Europe have demonstrated the process, equipment, and containment reliability. The gas centrifuges used in the NEF, Urenco's Model TC-12, are designed to operate continuously for many years. The resultant loads from centrifuge failures are restrained by the casing and the floor mounting element (flomel). These components are designed so rotor debris does not penetrate the casing and the flomels do not break away from the floor. The inventory of UF_6 in each centrifuge and in a cascade is low. The UF_6 is contained by the outer casings that are housed within enclosures for thermal stability.

3.4.3.4 Interfaces

The Cascade System interfaces with the following systems and utilities.

A. UF_6 Feed System

B. Product Take-off System

C. Tails Take-off System

D. Contingency Dump System

E. Centrifuge Cooling Water Distribution System

F. Compressed Air System

G. Electrical System

H. Plant Control System.

3.4.3.5 Design and Safety Features

The Cascade System is designed and constructed to provide safe operation for plant personnel as well as the general public. Release of UF_6 to the atmosphere is minimized by:

A. All process piping, valves and vessels that contain UF_6 operate at subatmospheric pressure. Initial leaks would be inward to the system. Abnormal pressures caused by such leaks or process upsets are detected by strategically located pressure sensors and indicated by alarms. Appropriate actions are initiated by the process operator. At certain levels, the actions begin automatically. Actions to stop UF_6 flow, isolate equipment, or shutdown systems are accomplished to avoid the release of UF_6.

B. If a centrifuge fails, i.e., "crashes," it is isolated to prevent contamination from entering other parts of the cascade. Current sensors are provided to detect crashes.

C. If a process upset occurs (pressure or temperature), the cascade is dumped to the Tails Take-off System. If the Tails Take-off System is unavailable, the gasses are evacuated to the Contingency Dump System.

D. The centrifuge outer casing is the primary barrier to the escape of UF_6. The casing encloses the rotor and its component parts and maintains them under vacuum. The outer casing provides confinement of the UF_6 in the centrifuge. It also serves to contain parts or fragments potentially spinning off a centrifuge during a failure. It is reinforced at both ends to contain the heavier rotor end caps and end cap fragments and has design features to prevent end cap debris from impacting non-reinforced areas of the casing. Cascades are designed so that failed centrifuges can be left in place.

E. The floor mounting element (flomel) and the associated bolts for the centrifuges are designed to remain intact after a rotor failure to prevent the centrifuge casing from breaking away and damaging other centrifuges or injuring workers. The flomel consists of a concrete floor mounting element with threaded metal inserts for anchoring the centrifuge foot flange via bolts. The flomel in turn is securely cast in the concrete floor of the Cascade Hall.

3.4.3.6 Operating Limits

The Cascade System for each Cascade Hall is capable of producing a maximum of 545,000 SWU/year. The nominal capacity of each Cascade Hall is 500,000 SWU/yr. It is limited to a maximum final product assay of 5.0 $^{w}/_{o}$ ^{235}U.

3.4.3.7 Instrumentation

The process variables such as pressures, temperatures, valve positions and flowrates are automatically controlled. Deviations from the specified values are detected and indicated via a two or three level alarm signal. Normally at the first alarm level, the process operator has the ability to manipulate the process to restore it to normal. At the second and sometimes the third alarm level, automatic action is taken to provide system protection. For safety, system protection and operability sensors may be single, duplicate (one out of two action) or triplicate (one or two out of three action).

Each cascade is provided with two control systems. Under normal operating conditions one system carries out all of the required process control and protection logic; the second system provides a safety 'envelope' around the control system functionality. The failsafe mode for both systems is Contingency Dump.

If any out-of-limit temperatures, pressures or cooling water temperatures are detected, the cascade is automatically shutdown and UF_6 evacuation to the Tails Take-off System is initiated.

3.4.3.8 Criticality Safety

3.4.3.8.1 Centrifuges and Cascades

Criticality safety of TC-12 centrifuges was assessed assuming 6 $^{w}/_{o}$ ^{235}U enrichment. The only potential for a criticality incident in a centrifuge cascade is by gross uranium accumulation in failed centrifuges. To achieve criticality in a cascade would require an array of failed centrifuges to be completely filled with uranic breakdown (as $UO_2F_2 \cdot 3.5H_2O$). The extreme conditions required to obtain the necessary uranic accumulation for criticality by this mechanism could never credibly occur in practice. Furthermore, it is highly unlikely that: (1) the centrifuges in such an array would fail simultaneously, (2) the failures would lead to inleakage of moist air into the failed centrifuges, (3) all the failed centrifuges would fill up with UF_6 breakdown products, and (4) would have an H/U ratio that is near optimum. Therefore, the possibility of a criticality incident in a centrifuge cascade can be considered not credible.

3.4.3.8.2 UF_6 Product Pipework

Product pipework in the Separations Building varies in size up to a maximum nominal diameter of 150 mm (5.9 in). Only minimal surface deposition of UF_6 occurs in pipework but criticality safety has been assessed for the possibility of localized blockages in pipes with the formation of uranyl fluoride due to air inleakage.

MONK8A (SA, 2001) calculations have been performed for generic arrays of pipe intersections, filled entirely with uranyl fluoride / water mixture at optimum moderation at 6.0 $^{w}/_{o}$ enrichment. The minimum permitted free space between intersections was determined to be 520 mm (20.5 in) for 150 mm (5.9 in) nominal pipe, and 135 mm (5.3 in) for 100 mm (3.9 in) nominal pipe; no spacing restriction applies to pipework of nominal diameter 65 mm (2.6 in).

The above restrictions apply to individual pipe runs with up to 64 intersections or adjacent pipe runs totaling up to 64 intersections.

Parallel pipe runs containing product material will either fit within the criticality safe value for cylinder diameter or be explicitly modeled.

The Separations Building pipework conforms to the above specifications. If not, explicit calculations will be performed. For example, the spacing restriction might not be satisfied, but the pipework might have fewer than 64 intersections.

3.4.4 Product Take-off System

The NEF Product Take-off System uses a process similar to the original Louisiana Energy Services Claiborne Enrichment Center, however there are differences. The NRC staff previously reviewed the Claiborne Enrichment Center license application relative to the Product Take-off System and concluded that the description, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion on the Product Take-off System is provided in NUREG-1491 (NRC, 1994), Section 3.5. The primary differences are:

Product Take-off Cylinder Operating Temperature

The Claiborne Enrichment Center cylinder temperature was maintained at +10°C (50°F). Cool air from a central system was used to maintain the temperature. The NEF chills cylinders to –25°C (-13°F) by using cold air from refrigeration units mounted on each LTTS.

System Pressure (at the product cylinder)

The Claiborne Enrichment Center used a relatively high pressure of 430 mbar (6.24 psia) in the header at the cylinder. The high pressure was generated via two pump sets. The first pump set was in the "primary" header from each cascade and consisted of two pumps – a first stage and a second stage - that were in series. There were seven cascades; therefore, there were 14 pumps total in the seven sets. After these seven pump sets, the discharges all combined into a single "secondary" header. In this secondary header, there were two high-pressure vacuum pumps. These two pumps were in parallel.

The pressure (vacuum) at the cylinder for the NEF is substantially lower. It has been reduced to no greater than 80 mbar (32.1 in. H_2O). This lower vacuum level is possible primarily because the cylinder is chilled to –25°C (-13°F). The product pumping system for the NEF combines the product from eight cascades into a main header and uses two vacuum pumps in series for each Cascade Hall. There is a spare set of vacuum pumps for each Cascade Hall. These are in parallel arrangement. The lower operating vacuum level eliminates the need for a high-pressure pump in the system.

Product Header Heat Tracing

The operating pressure in the Claiborne Enrichment Center header following the high-pressure vacuum pumps required heat tracing and valve hot boxes to prevent desublimation at the building temperatures. For the lower pressure in the NEF system, the building ambient temperature is sufficient to prevent desublimation and heat tracing is not necessary.

Product Vent Subsystem

The current system has two parallel UF_6 cold trap and vacuum pump/chemical trap sets for each Cascade Hall. The Claiborne Enrichment Center used three UF_6 cold traps and vacuum pump/chemical trap sets for each Separations Building Module, with a common spare shared between the two Cascade Halls.

3.4.4.1 Functional Description

The primary function of the Product Take-off System is to provide continuous withdrawal of the enriched gaseous UF_6 product from the centrifuge cascades. The product is transported via a train of vacuum pumps to chilled 30 or 48-in diameter cylinders where the UF_6 is desublimed. A secondary function of this system is to provide a means for venting light gas impurities from the enrichment process. The system is shown in Figure 3.4-10, Process Flow Diagram Product Take-off System.

Under normal operating conditions, the system produces small intermittent quantities of gaseous effluent from the treatment of light gas impurities in the Product Vent Subsystem. Additional small quantities of intermittent gaseous effluent are produced from purging and evacuating the flexible piping used to connect the product cylinders to the system during cylinder changeout. This effluent from the Product Vent Subsystem is routed to the Separations Building GEVS for further treatment. Solid wastes are produced from periodic change-out of chemical and oil traps. There is no liquid effluent directly produced in this system. Vacuum pumps are taken out of service for maintenance and the pump oil is reprocessed in the TSB and reused.

The Product Take-off System is located in the UF_6 Handling Area and the Process Services Area of the Separations Building. The major equipment locations are shown on Figure 3.3-2, Separations Building Module, First Floor; Figure 3.3-3, UF_6 Handling Area Equipment Location; and Figure 3.3-4, Separations Building Module, Second Floor. It is operated from the Control Room, with the exception of vacuum pump and cylinder maintenance and preparation operations, which are controlled locally.

3.4.4.2 Major Components

The major components of the Product Take-off System are listed below.

A. Product System Main Header

The product system main header connects each cascade to the product pumping trains. Pressure transducers in the header protect the cascades from air ingress or back flow of UF_6.

B. Product Pumping Trains

Each Cascade Hall has two product pumping trains connected in parallel. One pump train is on-line while the other is in standby or maintenance. Each train consists of a set of two vacuum pumps connected in series. Manual and automatic valves isolate each pump set. The pump train transports the UF_6 product from each cascade to the Product Low Temperature Take-off Stations.

C. Product Low Temperature Take-off Stations

The Product Low Temperature Take-off Station (LTTS) consists of a composite-wall insulated box. The Product LTTS panels have a non-flammable insulated core, and are vapor sealed to prevent ice build-up within the insulation. The Product LTTS is designed to prevent ice build-up within the box. The Product LTTS totally encloses the cylinder, cylinder support structure, and rails. The front of the Product LTTS has a single door through which the cylinder is inserted and removed. The back of the Product LTTS has an opening through which the cylinder is connected to the UF_6 piping. A rubber bellows is fitted around the back opening, which envelops the cylinder valve, to prevent cooled air from leaking out of the Product LTTS. A hot air blower is used to keep the valve and its surrounding area heated. The door frames, access port, rubber collar, and defrost condensate piping are provided with heat tracing to prevent ice build-up.

Each Product LTTS has a chiller unit, which is mounted on the top of the Product LTTS. This unit provides the cold air necessary to decrease the temperature in the box sufficiently to remove the thermal energy from the UF_6 gas and cause it to desublime in the cylinder. The chiller unit has a defrost cycle to remove ice from the cooling coils. This is done with a defrost heater at the coils.

The valves used to route the product to the appropriate Product LTTS, or for venting and purging, are mounted in a valve frame near each Product LTTS.

Each Product LTTS is provided with a weighing system, which incorporates a weigh frame, four load cells, and associated weighing instrumentation. The weigh system provides continuous measurement of the mass of UF_6 accumulating in the product cylinder.

D. Product Vent Subsystem

The Product Vent Subsystem consists of a product vent transfer header, two horizontal UF_6 cold traps, two heater/chiller units, two automatic control valves, and two vacuum pump/chemical trap sets. These components are discussed below.

1. UF_6 Cold Traps with Heater/Chiller Units.

Each UF_6 cold trap consists of an insulated horizontal tube with internal baffles and a dedicated heater/chiller unit. Each heater/chiller unit contains approximately 70 L (19 gal) of silicon oil, as the heat exchange media, which circulates around each cold trap. These Product Vent Subsystem heater/chiller units are separated by over 30 m (100 ft) from other heater/chiller units in similar subsystems. The UF_6 cold trap is chilled to cause any UF_6 in the vent gases to desublime. It is heated to sublime the trapped UF_6 for transfer back to a product cylinder. Each end of the UF_6 cold trap is heat traced to prevent the UF_6 from desubliming and blocking the inlet and outlet. The heat tracing also prevents ice from building up on the outside of the UF_6 cold trap and affecting the weighing system.

Each UF_6 cold trap is provided with a weighing system, which incorporates a weigh frame, four load cells, and associated weighing instrumentation. The weigh system provides continuous measurement of the mass of UF_6 accumulating in the UF_6 cold trap and indicates when it is full to prevent overfilling.

2. Vacuum Pump/Chemical Trap Sets.

The vacuum pump/chemical trap set consists of a carbon trap, an aluminum oxide trap, and an insulated vacuum pump with internal nitrogen purge and oil traps on either side. The exhaust from the pump goes to the Separations Building GEVS.

The activated carbon trap removes small traces of UF_6. The aluminum oxide trap removes HF. The oil traps are installed before the pump to prevent back diffusion and after the pump to prevent oil from being transferred into the Separations Building GEVS.

E. Assay Sampling System

Piping installed on the product header after the product pumping trains allows a product assay sample to be collected in a sample bottle. The sample system is comprised of automatic and manual valves, nitrogen purging, and an evacuation pump/chemical trap set similar to the one described above. However, this set does not contain an aluminum oxide trap for HF removal.

F. On-line Mass Spectrometer System

A piping connection on the product header, after the product pumping trains, allows a small gas sample to be fed to an on-line mass spectrometer. The analysis results allow any required adjustments to the cascades.

3.4.4.3 Design Description

The design bases and specifications are given in Table 3.4-4, Product Take-off System Design Basis. Applicable Codes and Standards are given in Table 3.4-5, Product Take-off System Codes and Standards.

The Product Take-off System is dedicated to an individual Cascade Hall of eight cascades. The system is designed to continuously remove the enriched UF_6 product from the cascades under all operating conditions. The maximum product flow rate of 18.4 kg (40.6 lb) per hour is based on a maximum capacity of 545,000 SWU per year (produced by each Cascade Hall).

The entire Product Take-off system operates at subatmospheric pressure. In the event of a containment failure (e.g., pipe leak), releases of UO_2F_2 and HF is greatly minimized because air would migrate into the system rather that UF_6 pouring out of the system. This important safety feature greatly limits the likelihood of exposures.

There are five Product Low Temperature Take-off Stations for each Cascade Hall. Of these five, two are on-line during normal operation. These two Product LTTSs are adequate to handle product flow when 30-in cylinders are being used. Two of the remaining three Product LTTSs are in standby auto. One of these Product LTTS is automatically switched to on-line when one of the two on-line cylinders is full. The fifth station is in standby (cylinder inside station but not on automatic), off-line, preparation (cylinder being removed or inserted), or maintenance mode.

Gaseous UF_6 product from the cascades flows from each centrifuge cascade, through the product main header, to the pumping trains. Typical main header pressures are on the order of a few mbar.

From the product pumping trains the UF_6 flows to the product cylinders housed in the Product LTTSs. The transfer header pressure is limited to 80 mbar (32.1 in. H_2O) to prevent UF_6 desublimation at ambient temperatures. Building ambient temperature is maintained above 18°C (64.4°F) so that heat tracing of the UF_6 transfer piping is not required.

Light gas impurities normally exit the centrifuges with the product rather than with the tails. To remove these impurities, the product cylinders are vented using a standby cylinder and the Product Vent Subsystem.

During production it is necessary to measure the concentration of the product or tails being produced. The operator can collect a sample for manual analysis using the Assay Sampling

System, or automatically measure the concentration using the On-line Mass Spectrometer System.

Materials of construction and fabrication specifications for the equipment and piping used in the Product Take-off System are compatible with UF_6 at the operating conditions and have been proven by over 30 years of use in existing Urenco European enrichment plants.

3.4.4.4 Interfaces

The Product Take-off System interfaces with the following systems and utilities.

A. Cascade System

B. Separations Building GEVS

C. Nitrogen System

D. Plant Control System

E. Compressed Air System

F. Electrical System

G. Hoisting and Transportation Equipment.

3.4.4.5 Design and Safety Features

This system is designed and constructed to provide safe operation for plant personnel as well as the general public. Principal design features are as follows:

A. All piping, vessels, and pumps in the Product Take-off System operate at subatmospheric UF_6 pressures.

B. Piping is all welded construction and process valves are bellows sealed.

C. Before carrying out any disconnections or connections of equipment, the piping is evacuated and purged with nitrogen. Flexible exhaust hoses connected to the Separations Building GEVS remove any releases from the work area.

D. Before discharge to the Separations Building GEVS, all gases flow across activated carbon and aluminum oxide to remove any traces of UF_6 and HF via the product vent vacuum pump/chemical trap set.

E. Temperature in each Product LTTS is monitored and controlled.

F. Product cylinder overfill is prevented by two weight trips. The first is at the desired net weight of UF_6 and the second is at the gross weight of the cylinder with UF_6 contents. Only the first trip is operator adjustable.

G. Removal of a connected cylinder from the Product LTTS is prevented by an interlock system. Unless the flexible hose on the cylinder valve has been removed and locked in its "holster," a physical barrier prevents the cylinder transporter drawbridge from docking with station rails, preventing cylinder removal.

H. Hydrocarbon lubricants are not used in any pumps. All pumps are lubricated with fully fluorinated synthetic oil such as "Fomblin," a perfluorinated polyether (PFPE).

I. Temperature and weight in the product vent vacuum pump/chemical trap set carbon trap are monitored and a trip on weight and a trip on temperature stops the product vent vacuum pump.

J. Mechanical interlocking systems are provided in all solid feed and low temperature stations to prevent the operation of the stations with an incorrect cylinder type loaded. The system prevents the use of 48 in cylinders identified for product take-off from being used in either a solid feed station or feed purification station.

3.4.4.6 Operating Limits

The Product Take-off System has the capacity to remove the UF_6 product on a continuous basis from the cascades at all rates under normal operating conditions. A Cascade Hall's normal maximum capacity is based on 545,000 SWU per year.

3.4.4.7 Instrumentation

The process variables, such as pressure, temperature, and valve position, are automatically controlled. Deviations from the specified values are detected and indicated by a two level alarm system. At the first alarm level, the process operator has the ability to manipulate the process to restore it to normal. At the second alarm level, automatic action is taken to provide system protection. For safety, system protection, and operability, sensors may be duplicated (one out of two action) or triplicated (one out of three action). Action is initiated if any one out of two or three sensors reach alarm levels.

A. Main Header

The product main header pressure is monitored with three pressure sensors. Normal operating pressure is less than 2 mbar (0.803 in H_2O). The first alarm level, high (H) is set to give operator warning of high pressure. A second alarm level, high high (HH) signals the Cascade System that the product main header is not available.

B. Product Pumping Trains

Each product pumping train inlet pressure is monitored. Normal operating pressure is less than 2 mbar (0.803 in H_2O). The first alarm level (H) warns the operator of high pressure. The second alarm level (HH) automatically closes the inlet and outlet valves and trips the pump train off-line to protect against air leakage into the cascades.

The outlet pressure of each product pumping train is monitored. Normal operating pressure is less than 55 mbar (22.1 in H_2O). The first alarm level, set at 70 mbar (28.1 in H_2O), provides the operator warning of high pressure. A second alarm level at 80 mbar (32.1 in H_2O) automatically closes the inlet and outlet valves and trips the pump train off-line.

C. Product Low Temperature Take-off Stations

Each product cylinder inlet pressure is monitored. Normal operating pressure is less than 50 mbar (20.1 in H_2O). The first alarm level is set at 50 mbar (20.1 in H_2O) to automatically initiate the timed cylinder venting sequence. A second alarm level set at 70 mbar (28.1 in H_2O) warns of high pressure. A third alarm level, at 80 mbar (32.1 in H_2O), closes the Product LTTS inlet valve and trips the Product LTTS off-line.

For weight control, each Product LTTS has a weighing system consisting of four load cells and a transmitter to monitor the contents of the product cylinder. A weight of less than 800 kg (1,764 lb) indicates no cylinder present in the Product LTTS. The first alarm, set at the net allowable weight of UF_6 in the product cylinder, promotes a standby Product LTTS to on-line and closes the Product LTTS inlet valve to prevent overfilling. A second alarm, set at the gross allowable weight of the product cylinder filled with UF_6, also closes the inlet valve and trips the Product LTTS off-line. The output of the weighing system also allows cylinder weight to be verified to be within specified trending limits.

For temperature control and protection from high temperatures, the Product LTTS has a stand-alone control and protection system. The total system consists of three sensors. For main Product LTTS temperature control, one sensor is mounted in the air return to the chiller unit and monitors the circulating air temperature. This sensor and local control maintains the Product LTTS temperature to a normal value of –25°C (-13°F). In addition to controlling the Product LTTS temperature, one output is monitored by the Plant Control System and warns when the air temperature rises to from -25°C (-13°F) to –5°C (23°F). This would indicate a chiller failure or that the defrost heater is not functioning properly. When the defrost heater is on, the circulating air fan is off to minimize the increase in Product LTTS air temperature. In addition to the closed loop control system previously described, there are two independent and diverse temperature protection instruments. These provide extra safety margin to protect against increases in temperature that may occur if the heater control did not operate properly. The first instrument measures the circulating air temperature and is fail-safe hardwired. The second measures the air inside the Product LTTS and is a fail-safe capillary device. Both of these instruments will trip the defrost heater and fan power supply in the event the air temperature rises above set points. Set point on the hardwired instrument is 50°C (122°F) and set point on the capillary instrument is 53°C (127°F). If heater trip occurs from these two instruments, the Product LTTS is automatically taken off-line and put into a standby mode.

To prevent desublimation in the cylinder valve, heated air is blown over the valve with a hot air blower. A temperature sensor on the valve controls the temperature to 63°C (145°F).

D. Product Vent Subsystem

1. UF_6 Cold Traps

The vent header pressure, between the Product LTTS and the UF_6 cold traps, is monitored. During the vent sequence the normal pressure is at or below 50 mbar (20.1 in. H_2O). During the gas-back sequence, when UF_6 is sublimed in the UF_6 cold trap for transfer back to a product cylinder, the header pressure is at the UF_6 vapor pressure. A gas-back first alarm level at 90 mbar (26.1 in. H_2O) warns of high pressure. A second alarm level at 99 mbar (39.7 in. H_2O) closes the Product LTTS vent valve to prevent flow back into the Product Take-off System.

During the venting operation, the product vent UF_6 cold trap outlet pressure is monitored. A first low alarm level at 20 mbar (8.03 in. H_2O) indicates the UF_6 cold trap is empty in gas back mode. A second low alarm level, at 1 mbar (0.401 in. H_2O), closes the UF_6 cold trap outlet valve automatically to prevent UF_6 flow to the vacuum pump. A first high alarm level at 70 mbar (28.1 in. H_2O) warns of high pressure. A second high alarm level, at 80 mbar (32.1 in. H_2O), switches the heater/chiller unit off, trips the UF_6 cold trap off-line, and closes the outlet valve.

A pressure sensor and control valve between each UF_6 cold trap and its vacuum pump/chemical trap set restricts the flow of light gases through the UF_6 cold trap to ensure all UF_6 desublimes and does not reach the carbon trap. The line pressure into the vacuum pump/chemical trap set is controlled at 3 mbar (1.2 in. H_2O).

A weighing system monitors the contents of the UF_6 cold trap. A first alarm at 20 kg (44.1 lb) warns that the UF_6 cold trap is approaching capacity. At 25 kg (55.1 lb) the UF_6 cold trap inlet and outlet valves are closed and the UF_6 cold trap is switched off-line.

The temperature of the UF_6 cold trap is controlled at –60°C (-76°F) during cooling to desublime any UF_6 and at 20°C (68°F) for heating during sublimation to empty the UF_6 cold trap of collected UF_6 (gas-back). A low alarm at –63°C (-81.4°F) warns of a chiller unit fault. A first high alarm at –52°C (-61.6°F) closes the UF_6 cold trap outlet valve and a second high alarm at 25°C (77°F) warns of high temperature during gasback. At 30°C (85°F) the unit trips off-line to avoid desublimation of UF_6 in the header.

2. Vacuum Pump/Chemical Trap Sets.

To prevent the carbon trap from overheating and overfilling with product, there are two instruments. One sensor monitors the chemical trap temperature. This sensor will close the product vent UF_6 cold trap outlet valve when carbon trap temperature exceeds 42°C (108°F). This blocks flow to the vacuum pump/chemical trap set. This sensor will also provide an automatic trip of the associated vacuum pump on carbon trip high temperature. The carbon trap also has a weigh system. In addition to local weight display, this system will shut down the vacuum pump when the high weight set point is reached.

The activated aluminum oxide (Al_2O_3) trap on the vacuum pump/chemical trap set is also equipped with a weigh system. The weigh system on the aluminum oxide trap only displays a weight locally. There is no control function on this weight indicator.

Increase in weight is used to monitor accumulation of UF_6 in the carbon trap and HF in the aluminum oxide trap. The traps are replaced based on the accumulated weight.

E. Assay Sampling Subsystem.

The assay sampling header pressure is monitored to prevent air entering the Product Take-off System and Tails Take-off System. A high level alarm at 70 mbar (28.1 in. H_2O) closes the assay sampling inlet valves. The sample inlet valves (product and tails) and the sample evacuation valve are interlocked, allowing only one of the valves to be open at any one time. Both sample inlet valve open cycles are timed.

3.4.4.8 Criticality Safety

3.4.4.8.1 Product Cylinders

The product enrichment within a 48Y or 30B product cylinder is limited to 5.0 $^{w}/_{o}$ ^{235}U by the plant design, configuration and operating features. The UF_6 content is limited to no more than the 48Y or 30B cylinder fill limit by the plant design and operating features. The moderation within the cylinder is controlled by a series of plant operating features. These features include, among others, checks that the cylinder is clean and empty prior to the commencement of fill. Also, the moderator (H_2O, HF) entering the cylinder is monitored during the time the cylinder is connected to the plant UF_6 systems.

Calculations were performed on infinite two-dimensional arrays of full 48Y or 30B product cylinders. Inside each cylinder a region of UO_2F_2/water mixture was located. The remainder of the interior of the cylinder was assumed to be filled with 6.0 $^{w}/_{o}$ ^{235}U enriched UF_6. Cylinders in the arrays were placed with the valve and base ends alternately in contact, so that the moderated region in a given cylinder was in the closest possible proximity to the moderated region in an adjacent cylinder. All cylinders were considered to be lying on a concrete pad one meter thick. Moderation was varied to obtain the optimum H/U ratio. Worst-case external reflection/moderation conditions were found by varying the density of the interstitial water between cylinders to simulate frost or snow. The calculation also assumed one cylinder above (touching) the array to simulate movement in/out/over the array.

For the 48Y cylinder, the condition that met the upper safety limit had an H/U ratio of 11.5 with an interstitial water density of 0.10 g/cm^3 (6.2 lb/ft^3). Thus, the maximum safe mass of hydrogen in each type product 48Y cylinder in an array was determined to be 1.05 kg (2.31 lb) present in the form of 9.5 kg (20.9 lb) of water.

For the 30B cylinder, the condition that met the upper safety limit had an H/U ratio of 10.5 with an interstitial water density of 0.25 g/cm^3 (15.6 lb/ft^3). Thus, the maximum safe mass of hydrogen in each type product 30B cylinder in an array was determined to be 0.95 kg (2.09 lb) present in the form of 8.5 kg (18.7 lb) of water.

Criticality safety of Type 48Y and 30B product cylinders depends on the control of moderator content. Criticality safety is achieved by ensuring that there is less than 1.05 kg (2.31 lb) of hydrogen present in a Type 48Y cylinder and less than 0.95 kg (2.09 lb) of hydrogen present in a Type 30B cylinder.

3.4.4.8.2 UF_6 Cold Traps

Although the cold traps have a large internal volume they are individually safe by shape, the trap body having an internal diameter of 20.3 cm (8.0 in). This compares with the safe diameter of 21.9 cm (8.6 in) for 6.0 $^{w}/_{o}$ enrichment. Individual cold traps are thus safe in isolation for any uranyl fluoride/water mixture. In practice the maximum H/U atom ratio in the cold traps will be 7; however, a sensitivity study is performed to determine the optimum H/U ratio, providing an additional margin of safety.

The cold trap and the standby cold trap are separated from each other by center-to-center separation of 110 cm (43.3 in). There is a minimum edge separation of 180 cm (70.9 in) from any other fixed plant vessels that can accumulate enriched uranium. The pair of traps can thus be considered to be neutronically isolated from other fixed vessels.

Calculations were performed on the isolated pair of cold traps and were found to be substantially subcritical with $k_{eff} = 0.8030$. The calculations assumed an enrichment of 6.0 $^{w}/_{o}$, H/U of 7 and 2.5 cm (0.984 in) water reflection placed at the model boundary to simulate spurious reflection.

According to the restrictions on movement of mobile vessels, one vessel can come into contact with a trap but any others have to be kept at 60 cm (23.6 in) separation.

MONK8A (SA, 2001) calculations have been performed in which a vacuum cleaner is in contact with one of the cold traps, and another vessel (a 14 L (3.7 gal) product vent vacuum pump) is at 60 cm (23.6 in) edge spacing from the same cold trap. These are typical of Separation Plant mobile vessels. Each mobile vessel was modeled with the appropriate uranic fill; the vacuum cleaner was filled with uranyl fluoride/water mixture with optimum moderation (H/U=12), and the vacuum pump (conservatively containing hydrocarbon oil) was filled with uranic breakdown of composition $UF_4 \cdot 10.5CH_2$. The resulting k_{eff} = 0.8229 shows a slight increase in reactivity with respect to the isolated pair of traps using the same conservative assumptions. The vacuum cleaner was assumed to be a cleaner of internal diameter 20.3 cm (8.0 in) and length 66 cm (26.0 in) and was assumed to be entirely filled with uranic material with an enrichment of 6.0 $^w/_o$. MONK8A (SA, 2001) calculations have been carried out for an isolated cylinder using these dimensions, filled with uranyl fluoride/water at optimum moderation and with 2.5 cm (0.984 in) water reflection. This gave a value for k_{eff} of 0.8037. The cleaner has high efficiency particulate air (HEPA) filtration on the exhaust, and will be dedicated for cleaning operations where uranic material is involved and will be marked clearly.

Additionally, calculations were performed in which it was assumed that there are no movement controls, and both the vacuum cleaner and pump were in contact with one of the cold traps. Even with 2.5 cm (0.984 in) spurious water reflection placed around each unit, and at enrichment of 6.0 $^w/_o$, the result remained substantially subcritical with k_{eff} = 0.8673.

The cold traps have therefore been determined to be safe both as a pair in isolation and while interacting with other fixed plant or vessels in movement for ^{235}U enrichments up to 6.0 $^w/_o$.

3.4.4.8.3 Vacuum Pump / Chemical Trap Sets

These chemical traps of the Product Vent Subsystem are individually safe by diameter (20.3 cm (8.0 in) compared with the safe diameter of 21.9 cm (8.6 in) calculated for 6.0 $^w/_o$ enrichment). However, calculations have been performed concerning the effect of possible neutron interaction with nearby (uranium bearing) equipment.

In the MONK8A (SA, 2001) calculations for the Product Vent Subsystem, the plant spacing to the edge of the standby vent system is assumed to be 50 cm (19.7 in). The standby vent system has been included in the model. The traps were both assumed to fill entirely with uranyl fluoride/water with no restriction on water content. This is conservative, as in practice the H/U ratio of the uranyl fluoride in the traps will have a limiting upper value of 7. Also, the space within the trap, which would normally be occupied by carbon or alumina, is modeled as being filled with uranic material. This maximizes the mass of fissile material within the traps and provides added conservatism. The pump, alumina traps, oil trap and exhaust filter are assumed to be filled with uranyl fluoride/water of unlimited water content. This is conservative, as virtually no uranium is expected in these components.

Calculations were performed to account for interaction with other vessels in movement. According to the restrictions on movement, one mobile vessel can come into contact with one of the fixed chemical absorber traps, but other mobile vessels are assumed to be at 60 cm (23.6 in) separation. The case modeled was for a vacuum cleaner (of diameter 20.3 cm (8.0 in) and length 66 cm (26.0 in)) to be brought into contact with the vacuum pump in the product vent array. One other item, a 14 L (3.7 gal) rotary vane pump, was placed at 60 cm (23.6 in) edge spacing from the vacuum cleaner. The vacuum cleaner was assumed to be a cleaner of internal diameter 20.3 cm (8.0 in) and length 66 cm (26.0 in) and was assumed to be entirely filled with uranic material with an enrichment of 6.0 $^{w}/_{o}$. MONK8A (SA, 2001) calculations have been carried out for an isolated cylinder using these dimensions, filled with uranyl fluoride/water at optimum moderation and with 2.5 cm (0.984 in) water reflection. This gave a value for k_{eff} of 0.8037. The cleaner has HEPA filtration on the exhaust, and will be dedicated for cleaning operations where uranic material is involved and will be marked clearly.

The MONK8A (SA, 2001) calculation for the worst case, where all vessels were assumed to be entirely filled with uranyl fluoride/water mixture at optimum moderation, a trap and a vacuum cleaner are in contact with one of the fixed pumps, and all pumps were modeled with volumes of 14 L (3.7 gal), yields a k_{eff} = 0.9328.

It should be noted that the above MONK8A (SA, 2001) model represents extreme accident conditions in terms of uranium accumulation and moderator ingress. It should also be noted that the simple MONK8A (SA, 2001) model used for the vacuum pump in all of the calculations is conservative. Since the real shape of the internal free volume is far from optimum, an explicit model of the pump is expected to result in a significant reduction in k_{eff}.

The vacuum pump/chemical trap sets have been shown to be safe under normal operating conditions and credible abnormal operating conditions, for ^{235}U enrichments up to 6.0 $^{w}/_{o}$.

3.4.4.8.4 Product Pumping Train UF_6 Pumps

More than 200 cm (78.7 in) separates each Product Pumping Train in the plant from other uranium containing vessels, so only interaction with mobile components needs be considered. Additionally, when being removed for repair or maintenance, a UF_6 pump might pass near to another similar pump.

The currently planned pump combination unit consists of two Leybold pumps, models WS2000 series and WS500 series, positioned in a fixed frame. The WS500 series has an internal free volume of 8.52 L (2.25 gal), which is less than half of the maximum safe volume of 18 L (4.8 gal) at 6.0 $^{w}/_{o}$ enrichment. Therefore the WS500 series pump can be modeled conservatively as an isometric cylinder of the same volume. However, the WS2000 series pump has an internal free volume of 33 L (8.7 gal), which considerably exceeds the safe volume, and even exceeds the minimum critical volume of 24 L (6.3 gal). Although the WS2000 series pump has a larger than critical internal free volume, the shape of the internal volume is far from the optimum. Therefore, the WS2000 pump was modeled in some detail based on drawings supplied by the manufacturer.

MONK8A (SA, 2001) calculations were initially performed for an isolated pump combination to assess the intrinsic safety of the combination. The maximum k_{eff} of 0.7479 was achieved using an enrichment of 6.0 $^{w}/_{o}$ and an optimum H/U ratio of 12. From this analysis, the pump combination in isolation can be regarded as being intrinsically safe. As mentioned above, there is potential for a second pump unit to approach when being removed for maintenance. Calculations were performed on pairs of pumps in contact with each other, either side by side, or touching at the gearbox ends. The most reactive case was with the gearbox ends touching (k_{eff} = 0.8277), assuming an enrichment of 6.0 $^{w}/_{o}$ and an optimum H/U ratio of 10.

To consider interaction of mobile vessels, calculations were performed which added a vacuum cleaner to the pair of pumps, either in contact with the gearbox end (with the pumps side by side) or alongside one of the pumps (with the pumps touching at the gearbox ends). The worst case was achieved with the latter arrangement giving a k_{eff} = 0.8444.

A 14 L (3.7 gal) isometric cylinder representing an additional pump in transit was then placed 60 cm (23.6 in) from the vacuum cleaner resulting in a k_{eff} = 0.8743. This increase reflects the fact that the 14 L (3.7 gal) pump is the most reactive unit in the array; over 80% of fission events occur inside the 14 L (3.7 gal) pump. The relative orientation of the product pumps and vacuum cleaner has little effect on the value of k_{eff} when the 14 L (3.7 gal) pump is present. The vacuum cleaner was assumed to be a cleaner of internal diameter 20.3 cm (8.0 in) and length 66 cm (26.0 in) and was assumed to be entirely filled with uranic material with an enrichment of 6.0 $^{w}/_{o}$. MONK8A (SA, 2001) calculations have been carried out for an isolated cylinder using these dimensions, filled with uranyl fluoride/water at optimum moderation and with 2.5 cm (0.984 in) water reflection. This gave a value for k_{eff} of 0.8037. The cleaner has HEPA filtration on the exhaust, and will be dedicated for cleaning operations where uranic material is involved and will be marked clearly.

Even assuming the most conservative geometry and moderation conditions, k_{eff} remains substantially subcritical. Note that the movement of vessels considered above is considered to be part of normal operating conditions. The abnormal operating condition pertaining to the vessels concerns the assumption that all the vessels are completely filled with uranic breakdown at optimum moderation. This would be extremely unlikely for a single vessel in the array, and even more unlikely for more than one vessel.

It can be concluded that:

- An array of two pump units is safe at any spacing. No restriction is placed on the moderator content of the pump units.
- One pump or pump unit may be moved, and may approach another similar pump unit or vacuum cleaner (of safe diameter) at any orientation, and without spacing restrictions. Other pumps (of 14 L (3.7 gal) internal volume or less) must not approach within 60 cm (23.6 in) of a product pumping train. No restriction is placed on the moderator content of any of the vessels.

3.4.5 Tails Take-off System

The NEF Tails Take-off System uses a process similar to the original LES plant. The NRC staff previously reviewed the Claiborne Enrichment Center license application relative to the Tails Take-off System and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion on the Tails Take-off System is provided in NUREG-1491 (NRC, 1994), Section 3.5. The primary differences are as follows:

A. Tails Take-off Cylinder Operating Temperature

The Claiborne Enrichment Center cylinder temperature was maintained at +3.9°C (39°F) by spraying the cylinders with chilled water. The NEF chills the cylinders to –25°C (-13°F) by using cold air from refrigeration units.

B. System Pressure (at the UBC)

The Claiborne Enrichment Center used a relatively high pressure of 225 mbar (3.26 psia) in the header to the cylinder. The high pressure was generated via two pump sets. The first pump set was in the "primary" header from each cascade and consisted of two pumps – a first stage and a second stage that were in series. There were seven cascades; therefore, there were 14 pumps total in the seven sets. After these seven pump sets, the discharges all combined into a single "secondary" header. In this secondary header, there were three high pressure vacuum pumps. These three were in parallel. The pressure (vacuum) at the cylinder for the NEF is substantially lower. It has been reduced to no greater than 80 mbar (32.1 in. H_2O). This lower vacuum is accomplished primarily because the cylinder is chilled to –25.0°C (–13°F). As with the Claiborne Enrichment Center, the tails pumping system for NEF uses two vacuum pumps in series for each cascade. There is a spare set of vacuum pumps for each cascade. These are in parallel arrangement. There is no high pressure pump in the secondary header.

C. Tails Evacuation Pump/Chemical Trap Set

The current system has a dedicated pump/chemical trap set for venting and does not use the Feed Purification System like the Claiborne Enrichment Center.

D. Cylinder Quantities

The Claiborne Enrichment Center contained a total of ten cylinders. There were five cooling stations, each with two cylinders. The NEF uses ten cylinders. However, each cylinder is in a dedicated LTTS.

3.4.5.1 Functional Description

The primary function of the Tails Take-off System is to provide continuous withdrawal of the gaseous UF_6 tails from the centrifuge cascades. The tails are transported via a train of vacuum pumps to 48-in diameter cylinders where the UF_6 gas is desublimed. A secondary function of this system is to provide a means for evacuating centrifuge cascades under abnormal operating conditions. The system is shown in Figure 3.4-11, Process Flow Diagram Tails Take-off System.

Most of the light gases from the separation process are discharged into the product stream, so venting of the tails system is seldom necessary.

Small, intermittent quantities of gaseous effluent are produced from purging and venting the flexible piping used to connect the UBCs to the system during cylinder changeout. This effluent is treated by the Tails Evacuation Pump/Chemical Trap Set to remove UF_6 or HF before being routed to the Separations Building GEVS for further treatment. Solid wastes are produced from periodic change-out of chemical and oil traps. There is no liquid effluent directly produced in this system. Vacuum pumps are taken out of service for maintenance and the pump oil is reprocessed in the TSB and reused.

The Tails Take-off System is located in the UF_6 Handling Area and Process Services Area of the Separations Building Module. The location of major equipment is shown on Figure 3.3-2, Separations Building Module, First Floor; Figure 3.3-3, UF_6 Handling Area, Equipment Location; and Figure 3.3-4, Separations Building Module, Second Floor. The equipment is operated from the Control Room with the exception of maintenance and preparation activities, which are controlled locally.

3.4.5.2 Major Components

The major components of the Tails Take-off System are:

A. Primary Header

The tails primary header connects each cascade to the Tails Pumping Trains. Pressure transducers in the header protect the cascades from air ingress.

B. Tails Pumping Trains

Each cascade has two dedicated Tails Pumping Trains connected in parallel. One pump train is on-line while the other is in standby. Each train has one set of pumps. Each set consists of two vacuum pumps in series mounted on a common frame. Manual and automatic valves isolate each pump set.

C. Secondary Header

Tails Pumping Trains discharge into the secondary header. The secondary header connects with the Tails Low Temperature Take-off Stations.

D. Tails Low Temperature Take-off Stations (LTTS)

The Tails LTTS consists of a composite-wall insulated box. The Tails LTTS panels have a non-flammable insulated core, and are vapor sealed to prevent ice build-up within the insulation. The Tails LTTS is designed to prevent ice build-up within the Tails LTTS. The Tails LTTS totally encloses the cylinder, cylinder support structure, and rails. The front of the Tails LTTS has a single door through which the cylinder is inserted and removed. The back of the Tails LTTS has an opening through which the cylinder is connected to the UF_6 piping. A rubber bellows is fitted around the back opening, which envelops the cylinder valve, to prevent cooled air from leaking out of the Tails LTTS. A hot air blower is used to keep the valve and its surrounding area heated. The door frames, access port, rubber collar, and defrost condensate pipework are provided with heat tracing to prevent ice build-up.

Each Tails LTTS has a chiller unit, which is mounted on the top of the Tails LTTS. This unit provides the cold air necessary to decrease the temperature in the box sufficiently to remove the thermal energy from the UF_6 gas and cause it to desublime in the cylinder. The chiller unit has a defrost cycle to remove ice from the cooling coils. This is done with a defrost heater at the coils.

The valves between the secondary header and the Tails LTTS are mounted in separate frames that are not attached to the Tails LTTS; however, they are in close proximity.

Each Tails LTTS is provided with a weighing system which incorporates a weigh frame, four load cells, and associated weighing instrumentation. The weigh system provides continuous measurement of the mass of UF_6 accumulating in the UBC.

E. Tails Evacuation Pump/Chemical Trap Set

The Tails Evacuation Pump/Chemical Trap Set consists of a carbon trap, an aluminum oxide trap, and an insulated vacuum pump with internal nitrogen purge and oil traps on either side. The exhaust from the pump goes to the Separations Plant GEVS.

The activated carbon trap removes small traces of UF_6. The aluminum oxide trap removes HF. Oil traps are installed before and after the pump to prevent oil migration both upstream and into the Separations Plant GEVS.

F. Assay Sampling Subsystem

Pipework is installed in the secondary header for sampling. The tails assay sample is taken into sample bottles at this point. The sample system is comprised of automatic and manual valves, nitrogen purging, and an evacuation pump/chemical trap set similar to the one described above.

G. On-line Mass Spectrometer System

Piping is installed in the secondary header to allow a small gas sample to be fed to an on-line mass spectrometer. The results of the mass spectrometer analysis are used to make process adjustments to the cascades.

3.4.5.3 Design Description

The design bases and specifications are given in Table 3.4-6, Tails Take-off System Design Basis. Applicable Codes and Standards are given in Table 3.4-7, Tails Take-off System Codes and Standards.

The Tails Take-off System is dedicated to an individual Cascade Hall consisting of eight cascades. The system is designed to continuously remove depleted UF_6 (tails) from the cascades under all operating conditions. The maximum tails flow is 168 kg/hr (370 lb/hr) based on a maximum capacity of 545,000 SWU/year (produced by each Cascade Hall). Peak flow rates could be as high as 256 kg/hr (564 lb/hr) for UF_6 removal from the cascades under abnormal conditions.

The entire Tails Take-off System operates at subatmospheric pressure. In the event of a confinement barrier failure (e.g., pipe leak), releases of UO_2F_2 and HF is greatly minimized because air would migrate into the system rather that UF_6 exiting the system. This important safety feature greatly limits the likelihood of worker and public exposures.

There are ten Tails LTTSs for each Cascade Hall. Of these ten, seven are on-line during normal operation. These seven are adequate for normal operations as well as peak flows generated during a cascade trip. One Tails LTTS is in standby auto. This Tails LTTS is automatically switched to on-line when one of the seven on-line cylinders is full. The other two Tails LTTS are in either standby manual (cylinder inside station but not on automatic), off-line, preparation (cylinder being removed or inserted), or maintenance mode.

Gaseous UF_6 tails from the cascades flows from each centrifuge cascade, through the primary header, to the tails pumping trains. Typical primary header pressures are of the order of a few mbar (in. H_2O).

From the tails pumping trains the UF_6 flows through the secondary header to the UBCs housed in the Tails LTTSs. The secondary header pressure is limited to 80 mbar (32.1 in. H_2O) to prevent UF_6 desublimation at ambient temperatures. Building ambient temperature is maintained above 18°C (64.4°F) so that heat tracing of the UF_6 piping is not required.

All components of the Tails Take-off System operate at subatmospheric pressure. Release of UF_6 and/or HF is unlikely because leakage, if it were to occur, would be inward to the system.

Materials of construction and fabrication specifications for the equipment and piping used in the Tails Take-off System are compatible with UF_6 at the operating conditions and have been proven by over 30 years of use in existing Urenco European enrichment plants.

3.4.5.4 Interfaces

The Tails Take-off System interfaces with the following systems and utilities:

A. Cascade System

B. Plant Control System

C. Nitrogen System

D. Compressed Air System

E. Separations Building GEVS

F. Electrical System

G. Hoisting and Transportation Equipment.

3.4.5.5 Design and Safety Features

This system is designed and constructed to provide safe operation for plant personnel as well as the general public. Principal design features are as follows.

A. All piping, vessels, and pumps in the Tails Take-off System operate at subatmospheric UF_6 pressures.

B. Piping is all welded construction and process valves are bellows sealed.

C. Before carrying out any disconnections or connections of equipment, the piping is evacuated and purged with nitrogen. Flexible exhaust hoses connected to the Separations Building GEVS remove any releases from the work area.

D. Before discharge to the Separations Building GEVS, all gases flow across activated carbon and aluminum oxide to remove any traces of UF_6 and HF via the Tails Evacuation Pump/Chemical Trap Set.

E. Temperature in each Tails LTTS is monitored and controlled.

F. Cylinder overfill is prevented by two weight trips. The first is at the desired net weight of UF_6 and the second is at the gross weight of the cylinder with UF_6 contents. Only the first trip is operator adjustable.

G. Removal of a connected cylinder from the Tails LTTS is prevented by an interlock system. Unless the flexible hose on the cylinder valve has been removed and locked in its "holster," a physical barrier prevents the cylinder transporter drawbridge from docking with station rails, preventing cylinder removal.

H. Hydrocarbon lubricants are not used in any pumps. All tails pumps are lubricated with fully fluorinated synthetic oil such as "Fomblin," a perfluorinated polyether (PFPE).

I. Temperature in the Tails Evacuation Pump/Chemical Trap Set carbon trap is monitored and controlled.

3.4.5.6 Operating Limits

The Tails Take-off System will have the capacity to remove the UF_6 tails on a continuous basis from the cascades at all rates under normal operating conditions. A Cascade Hall's normal maximum capacity is based on 545,000 SWU/yr. The system will also have the capacity to evacuate the full flow of UF_6 from the cascades under abnormal operating conditions.

3.4.5.7 Instrumentation

The process variables such as pressure, temperature, and valve positions, are automatically controlled. Deviations from the specified values are detected and indicated via a two level alarm system. At the first alarm level, the process operator has the ability to manipulate the process to restore it to normal. At the second alarm level, automatic action is taken to provide system protection. For safety, system protection and operability, sensors may be installed in duplicate (one out of two action) or triplicate (two out of three action). Action is initiated if any one out of two (or two out of three) sensor reaches alarm levels.

A. Primary Header.

There are two pressure transducers in each of the tails primary headers. Normal pressure is less than 2 mbar (0.8 in. H_2O). First alarm level (H) is a high level to give operator warning of high pressure. Second alarm level (HH) signals that the tails system is unavailable, to protect the cascade from high pressure.

B. Tails Pumping Trains.

Each Tails Pumping Train inlet pressure is monitored. Normal pressure is less than 4 mbar (0.8 in. H_2O). First alarm level (H) gives operator warning of high pressure. Second alarm level (HH) trips the vacuum pump off-line to protect the cascade from air ingress. A third alarm at 80 mbar prevents the pump from running and the outlet valve from opening to protect against gross leakage into the system.

C. Secondary Header.

The tails secondary pipe header pressure is monitored with three sensors. Normal pressure is less than 55 mbar (22.1 in. H_2O). The first alarm level provides operator warning of high pressure at 70 mbar (28.1 in. H_2O). At the second alarm level, 80 mbar (32.1 in. H_2O) on two of three sensors, the vacuum pump trips off-line and a signal that the tails system is unavailable goes to the programmable logic controller (PLC) in each cascade.

D. Tails Low Temperature Take-off Stations.

For pressure control, each tails cylinder inlet pressure is monitored. Normal pressure is between 5 and 50 mbar (2 and 20 in H_2O). The first alarm level is 70 mbar (28.1 in H_2O) to give operator warning of high pressure. The second alarm level at 80 mbar (32.1 in H_2O) automatically closes the Tails LTTS inlet valve and trips the Tails LTTS off-line.

For weight control, each Tails LTTS has a weighing system consisting of four load cells and a transmitter to monitor the contents of the UBCs. A weight of less than 800 kg (1,764 lb) indicates no cylinder present in the Tails LTTS. The first alarm, set at the net allowable weight of UF_6 for the 48-in cylinder, trips the Tails LTTS to standby to prevent overfilling. This promotes the standby auto Tails LTTS to on-line. The second trip, set at the gross allowable weight of a 48-in cylinder filled with UF_6, closes the inlet valve and trips the Tails LTTS to off-line. The output of the weighing system also allows cylinder weight to be verified to be within specified trending limits.

For temperature control and protection from high temperatures, the Tails LTTS has a stand-alone control and protection system. The total system consists of three sensors. For main Tails LTTS temperature control, one sensor is mounted in the air return to the chiller unit and monitors the circulating air temperature. This sensor and local control maintains the Tails LTTS temperature to a normal value of –25°C (-13°F). In addition to controlling the station temperature, one output is monitored by the Plant Control System (PCS) and warns when the air temperature rises to –5°C from –25°C (23°F from -13°F). This would indicate a chiller failure or that the defrost heater is not functioning properly. When the defrost heater is on, the circulating air fan is off to minimize the increase in Tails LTTS air temperature.

In addition to the closed loop control system previously described, there are two independent and diverse temperature protection instruments. These provide extra safety margin to protect against increases in temperature that may occur if the heater control does not operate properly. The first instrument measures the circulating air temperature and is fail-safe hardwired. The second measures the air inside the Tails LTTS and is a fail-safe capillary device. Both of these instruments will trip the defrost heater and fan power supply in the event the air temperature rises above set points. Set point on the hardwired instrument is 50°C (122°F) and set point on the capillary instrument is 53°C (127°F). If heater trip occurs from these two instruments, the Tails LTTS is automatically taken off-line and put into a standby mode.

To prevent desublimation in the cylinder valve, hot air is blown over the valve with a hot air blower. A temperature sensor on the valve controls the temperature to 63°C (145°F).

E. Tails Evacuation Pump/Chemical Trap Set

To prevent the carbon trap from overheating and overfilling with UF_6, there are two instruments. One sensor monitors the carbon trap temperature. This sensor will close the Tails LTTS vent valve when carbon trap temperature exceeds 42°C (108°F). This blocks flow to the vacuum pump/chemical trap set. The carbon trap also has a weigh system. In addition to local weight display, this system will shut down the vacuum pump when the high weight set point is reached.

The activated aluminum oxide (Al_2O_3) trap on the vacuum pump/chemical trap set is also equipped with a weigh system. The weigh system on the aluminum oxide trap only displays a weight locally. There is no control function on this weight indicator.

Increase in weight is used to monitor accumulation of UF_6 in the carbon trap and HF in the aluminum oxide trap. The chemical traps are replaced based on the accumulated weight.

3.4.6 Product Blending System

The NEF Product Blending System uses a process similar to the original LES plant. The NRC staff previously reviewed the Claiborne Enrichment Center SAR application relative to the Product Blending System and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion on the Product Blending System is provided in NUREG-1491 (NRC, 1994), Section 3.6. The primary differences are as follows:

A. Blending Donor Station Operating Conditions.

The Claiborne Enrichment Center used a Donor Station that operated above atmospheric pressure. UF_6 in the donor cylinder was maintained in the liquid phase. Normal UF_6 pressure in the feed cylinder was above atmospheric, at 2.5 bar (36.3 psia). Normal station heating temperature was up to 110°C (230°F). The Claiborne Enrichment Center used a sealed autoclave for secondary confinement of the donor cylinder to prevent exposure in the event a leak developed in the primary confinement barrier (cylinder and piping).

The NEF sublimes solid UF_6 directly to gaseous UF_6 at subatmospheric pressure, without entering the liquid phase. Normal donor cylinder pressure is 500 mbar (7.25 psia) and the station temperature during heating is limited to 61°C (142°F). As a result, a Blending Donor Station is used to heat the donor cylinder rather than an autoclave.

B. Blending Receiver Station Operating Temperature.

The Claiborne Enrichment Center cylinder temperature was maintained at +10°C (50°F). Cool air from a central system was used to maintain the temperature of the receiver stations. The NEF will chill the cylinder to –25°C (-13°F) by using cold air from a refrigeration unit integral to the Blending Receiver Station.

Other differences are the use of only four receiver stations in this process versus five in the original and the use of a dedicated vacuum pump/chemical trap set in the current design versus a mobile set in the original.

3.4.6.1 Functional Description

The primary function of the Product Blending System is to provide a means to fill 30B cylinders with UF_6 at a specified ^{235}U concentration. This is achieved by either transferring product from one donor cylinder into one receiver cylinder or blending product from multiple donor cylinders into one or more receiver cylinders. The system is shown in Figure 3.4-12, Process Flow Diagram Product Blending System.

Small intermittent quantities of gaseous effluent are produced from purging and evacuation of flexible piping during connection and removal of both donor and receiver cylinders. The effluent is treated in the Blending and Sampling Vent Subsystem to remove UF_6 and HF, and then discharged to the Separations Building GEVS for further treatment. Solid effluents are produced from periodic change-out of chemical and oil traps. There are no liquid effluents directly produced in this system. When the Blending and Sampling Vent Subsystem vacuum pump is taken out of service for maintenance, the oil is reprocessed in the TSB for reuse.

The Product Blending System is located in the Blending and Liquid Sampling Area of the Separations Building. The location of major equipment is shown on Figure 3.3-10, Cylinder Receipt and Dispatch Building, First Floor, Part A. It is operated from the Control Room, with the exception of preparation and maintenance activities that are performed locally at the equipment.

3.4.6.2 Major Components

The major components of the Product Blending System are listed below:

A. Blending Donor Station

A Blending Donor Station consists of an insulated box with a non-flammable insulated core. Each Blending Donor Station includes an electrical air heater and circulation fan to provide the thermal energy to sublime the solid UF_6 in the cylinder.

A weighing system is provided in the Blending Donor Station that consists of a weigh frame with four load cells. This system is used to provide continuous on-line weighing of the donor cylinder to monitor the quantity of UF_6. The weighing system is also used to indicate when the cylinder has transferred the required quantity of UF_6 and automatically close the Blending Donor Station outlet valve.

B. Donor Station Valve Hotbox

Valves in a Donor Station Valve Hotbox connect the donor cylinder to its Transfer Header, the Blending and Sampling Vent Subsystem, or the Nitrogen System. Manual and automatic isolation valves and pressure transducers are contained in the electrically heated Donor Station Valve Hotboxes to maintain them at a stable temperature. The UF_6 piping between the Blending Donor Station and Donor Station Valve Hotbox is heat traced.

C. Blending Transfer Headers

To provide operating flexibility there are two transfer headers that are used for transferring UF_6 from Blending Donor Stations to Blending Receiver Stations. Both UF_6 transfer headers are heat traced. In addition a vent header connects all the Blending Donor Stations and Blending Receiver Stations to the Blending and Sampling Vent Subsystem. The transfer headers are arranged such that a number of blending or transfer operations can take place at the same time.

D. Blending Receiver Station

A Blending Receiver Station consists of a composite panel box construction complete with rails for the electric carriage of the cylinder transporter. The Blending Receiver Station panels have a non-flammable insulated core and are vapor sealed to prevent ice build-up within the insulation. Each Blending Receiver Station incorporates an air chiller unit, with controls, to remove thermal energy from the UF_6 gas to cause it to desublime in the cylinder. The chiller unit has a defrost cycle, using a heater, to prevent ice buildup on the coils. A hot air blower directed at the cylinder valve prevents UF_6 from desubliming and blocking the cylinder inlet. A weighing device is provided in the Blending Receiver Station (a frame with four load cells and associated instrumentation) to provide continuous on-line weighing of UF_6 in the receiver cylinder to prevent overfilling.

The front of the Blending Receiver Station is made up of a single door and the back is furnished with an opening to facilitate connection of the cylinder to the UF_6 piping. A rubber bellows is fitted around the back opening, which envelops the cylinder valve, to prevent cooled air from leaking out of the Blending Receiver Station. Similar seals on the other openings in the Blending Receiver Station minimize leaks for energy conservation. The Blending Receiver Station access openings are provided with heat tracing to prevent ice build-up.

E. Receiver Station Valve Hotbox

Valves in the Receiver Station Valve Hotbox connect the Blending Receiver Station to both UF_6 Transfer Headers, the Blending and Sampling Vent Subsystem, or the Nitrogen System. Manual and automatic isolation valves and a pressure transducer are contained in the electrically heated Receiver Station Valve Hotbox to maintain them at a stable temperature.

The UF_6 piping between the Receiver Station Valve Hotbox and the Blending Receiver Station is heat traced.

F. Blending and Sampling Vent Subsystem

The Blending and Sampling Vent Subsystem consists of a UF_6 cold trap with its heating and cooling systems and a vacuum pump/chemical trap set. The Blending and Sampling Vent Subsystem serves both the Product Blending System and the Product Sampling System. The Blending and Sampling Vent Subsystem contains the following major components.

1. UF_6 Cold Trap.

The UF_6 cold trap consists of an insulated horizontal tube with internal baffles. It also has a dedicated heater/chiller unit operating at a cooling set point and a heating set point. The heater/chiller unit contains approximately 70 L (19 gal) of silicon oil, as a heat exchange media, which circulates around the cold trap. The low temperature removes the thermal energy from the UF_6 gas, causing it to desublime on the internal walls of the UF_6 cold trap, while leaving the light gas in the gaseous phase. The high temperature results in sublimation of the UF_6 contents of the UF_6 cold trap for transfer back to a receiver cylinder. Each end of the UF_6 cold trap is heat traced to prevent the UF_6 from solidifying and blocking the UF_6 cold trap entrance or exit. The UF_6 cold trap has a weighing device to provide continuous on-line weighing of the UF_6 accumulated.

An automatic control valve located after the UF_6 cold trap restricts the flow of gases through the UF_6 cold trap. This ensures an adequate residence time for the gases in the UF_6 cold trap to allow all of the UF_6 to desublime.

The UF_6 cold trap also provides the capability for emptying sample bottles, using a small manifold located upstream of the UF_6 cold trap. The temperature difference of the sample bottle at ambient and the UF_6 cold trap at –60°C (-76°F) allows the UF_6 to outgas without heating the bottle.

2. Vacuum Pump/Chemical Trap Set.

The UF_6 cold trap is followed by a vacuum pump/chemical trap set. The set consists of a carbon trap, an aluminum oxide trap, an insulated vacuum pump with nitrogen purge, and an oil trap on either side of the pump. The pump exhausts into the Separations Building GEVS.

The activated carbon trap removes any traces of UF_6 not desublimed in the UF_6 cold trap. HF is removed from the gas flow by the aluminum oxide trap. These traps are installed in front of the vacuum pump. Weigh cells are installed on the carbon trap and the aluminum oxide trap to indicate the accumulated mass in each without the need to remove the trap for weighing. Oil traps are installed before and after the vacuum pump to prevent diffusion of oil, both back into the Blending and Sampling Vent System and forward into the Separations Building GEVS.

3.4.6.3 Design Description

The design bases and specifications are given in Table 3.4-8, Product Blending System Design Basis. Applicable codes and standards are given in Table 3.4-9, Product Blending System Codes and Standards.

The Product Blending System is sized for the complete 3,000,000 SWU per year enrichment plant capacity. Gaseous UF_6 is transferred from the Blending Donor Stations to the Blending Receiver Stations through a system of valves and transfer headers.

The entire Product Blending System operates at subatmospheric pressure. In the event of a confinement barrier failure (e.g., pipe leak), releases of UO_2F_2 and HF are greatly minimized because air would migrate into the system rather that UF_6 exiting the system. This important safety feature greatly limits the likelihood of worker and public exposures.

There are two Blending Donor Stations with valve hotboxes, each connected to one of the two transfer headers in the Product Blending System. At any time one or both stations, each connected to a different header, can be on-line to handle the various blending or transfer operations.

There are four Blending Receiver Stations, each with a valve hotbox, connected in parallel to the two transfer headers. Any number of Blending Receiver Stations can be connected to a single header at any one time, but a single Blending Receiver Station cannot be connected to both headers at the same time.

The pressure in each UF_6 transfer header is limited to 500 mbar (7.25 psia). To prevent UF_6 desublimation at ambient building temperatures, the headers are heat traced. Building ambient temperature is maintained above 18°C (64.4°F).

All components and piping in the Product Blending System operate at subatmospheric pressure. Release of UF_6 and/or HF is unlikely because leakage, if it were to occur, would be into the system.

Materials of construction and fabrication specifications for the equipment and piping used in the Product Blending System are compatible with UF_6 at the operating conditions and have been proven by over 30 years of use in existing Urenco European enrichment plants.

3.4.6.4 Interfaces

The Product Blending System interfaces with the following systems and utilities.

A. Separations Building GEVS

B. Plant Control System

C. Nitrogen System

D. Compressed Air System

E. Electrical System

F. Hoisting and Transportation Equipment.

3.4.6.5 Design and Safety Features

The Product Blending System is designed and constructed to provide safe operation for plant personnel as well as the general public. Principal design features are as follows:

A. All process piping, valves, vessels and pumps in the Product Blending System operate at subatmospheric pressure.

B. Piping is all welded construction and process valves are bellows sealed.

C. Before disconnecting any equipment, the process piping is evacuated and purged with nitrogen.

D. A local exhaust to the Separations Building GEVS is provided any time a UF_6 line is disconnected.

E. Before discharge to the Separations Building GEVS, all gases flow across activated carbon and aluminum oxide in the Blending and Sampling Vent Subsystem chemical traps to remove any traces of UF_6 and HF.

F. Temperature in each Blending Donor Station and Blending Receiver Station is monitored and controlled.

G. Receiver cylinder overfill is prevented by two weight trips. The first is at the desired net weight of UF_6 and the second is at the gross weight of the cylinder with UF_6 contents. Only the first trip is operator adjustable.

H. Hydrocarbon lubricants are not used. The Blending and Sampling Vent Subsystem vacuum pump is lubricated with fully fluorinated synthetic oil such as "Fomblin," a perfluorinated polyether (PFPE).

I. Removal of a connected cylinder from a Blending Donor Station or a Blending Receiver Station is prevented by an interlock system. Unless the flexible hose on the cylinder valve has been removed and locked in its "holster," a physical barrier prevents the cylinder transporter drawbridge from docking with the station rails, preventing cylinder removal.

J. Temperature and weight in the Blending and Sampling Vent Subsystem carbon trap is monitored and a trip on weight and a trip on temperature stops the Blending and Sampling Vent vacuum pump.

K. Should a blockage occur in a section of process piping, the heat tracing on that section of pipe is not allowed to be switched on until the solid UF_6 has been removed.

3.4.6.6 Operating Limits

The Product Blending System is capable of handling the enrichment blending requirements of the entire plant. Since customers' enrichment requirements are generally met via adjustments to the enrichment process, blending is not always necessary.

3.4.6.7 Instrumentation

The process variables, such as pressures, temperatures and valve positions are automatically controlled. Deviations from the specified values are detected and indicated via two level alarm systems. At the first alarm level, the process operator has the ability to manipulate the process to restore it to normal. At the second alarm level, automatic action is taken to provide system protection. For safety, system protection, and operability, some sensors are duplicated. Action is initiated if any one out of two sensors reach alarm levels.

A. Blending Donor Station.

Both the Blending Donor Station air temperature and cylinder temperature are monitored to prevent over pressurization of the donor cylinder due to overheating. Normal air temperature in the Blending Donor Station during heating ranges from ambient to 61°C (142°F), while the cylinder temperature ranges from ambient to 53°C (127°F). The first alarm level is 62°C (144°F) for the Blending Donor Station air and 54°C (129°F) for the cylinder to give the operator warning of high temperature. The second alarm level is 55°C (131°F) for the cylinder, which trips the Blending Donor Station heater off.

In addition to the above temperature controls, the Blending Donor Station has two independent and diverse temperature protection instruments. One is hard wired and measures cylinder temperature, and the other is a capillary type and measures the Blending Donor Station air temperature. These provide extra safety margin to prevent overheating the cylinder if the air temperature control fails. Both systems automatically de-energize the air heater and blower, if either the cylinder temperature reaches 55°C (131°F) or the Blending Donor Station air temperature reaches 63°C (145°F).

The donor cylinder pressure is monitored with dual sensors to prevent over-pressurization. Normal header pressure is limited to 500 mbar (7.25 psia). The first alarm level is 600 mbar (8.7 psia) to give operator warning of high pressure. The second alarm level at 850 mbar (12.3 psia) automatically closes the cylinder valve and trips the Blending Donor Station off-line. A low pressure alarm at 200 mbar (2.9 psia) warns that a cylinder vent is complete.

Each Blending Donor Station has a weighing system to monitor the mass of UF_6 remaining in the cylinder. The first weight trip at 800 kg (1,764 lb) gross is used to indicate a cylinder is present in the Blending Donor Station. The second weight trip, equal to the net cylinder contents weight after meeting the receiver cylinder requirements, indicates that the target transfer weight has been reached and trips the Blending Donor Station to standby. A third weight trip signals that the donor cylinder is empty and trips the Blending Donor Station to standby.

B. Blending Receiver Station.

The weight of the receiver cylinder is monitored to determine when the required amount of UF_6 has been transferred and to protect against overfilling the cylinder. A low weight trip at 800 kg (1,764 lb) gross indicates that a cylinder is present in the Blending Receiver Station. The Blending Receiver Station trips to standby and automatically closes the inlet valve when the required transfer weight is reached. A second trip, at the maximum net weight for a 30B cylinder, also trips the Blending Receiver Station to standby and closes the inlet valve. A third trip, at the maximum gross weight for a 30B cylinder, closes the inlet valve and trips the Blending Receiver Station off-line. The output of the weighing system also allows cylinder weight to be verified to be within specified trending limits.

The receiver cylinder inlet pressure is monitored to assure that a cylinder is connected to the system. Normal pressure is from 0 to 500 mbar (0 to 7.25 psia). A first alarm level at 550 mbar (7.98 psia) warns the operator of high pressure. A second alarm level at 650 mbar (9.43 psia) automatically closes the Blending Receiver Station inlet valve and trips the Blending Receiver Station off-line.

For temperature control and protection from high temperatures, the Blending Receiver Station has a stand-alone control and protection system. The total system consists of three sensors. For main Blending Receiver Station temperature control, one sensor is mounted in the air return to the chiller unit and monitors the circulating air temperature. This sensor and local control maintains the Blending Receiver Station temperature to a normal value of –25°C (-13°F). In addition to controlling the Blending Receiver Station temperature, one output is monitored by the Plant Control System and warns when the air temperature rises to –5°C from –25°C (23° F from -13°F). This would indicate a chiller failure or that the defrost heater is not functioning properly. When the defrost heater is on, the circulating air fan is off to minimize the increase in Blending Receiver Station air temperature.

In addition to the closed loop control system previously described, there are two independent and diverse temperature protection instruments. These provide extra safety margin to protect against increases in temperature that may occur if the heater control does not operate properly. The first instrument measures the circulating air temperature and is fail-safe hardwired. The second measures the air inside the Blending Receiver Station and is a fail-safe capillary device. Both of these instruments will trip the defrost heater and fan power supply in the event the air temperature rises above set points. Set point on the hardwired instrument is 50°C (122°F) and set point on the capillary instrument is 53°C (127°F). If heater trip occurs from these two instruments, the Blending Receiver Station is automatically taken off-line and the transfer sequence stopped.

To prevent desublimation in the cylinder valve, hot air is blown over the valve with a hot air blower. A temperature sensor on the valve controls the temperature to 63°C (145°F).

C. Blending and Sampling Vent Subsystem UF_6 Cold Trap.

During the venting operation, the Blending and Sampling Vent Subsystem UF_6 cold trap outlet pressure is monitored. A first high alarm, at 70 mbar (28.1 in. H_2O), warns of high pressure in the UF_6 cold trap. A first low alarm, at 20 mbar (8.03 in. H_2O), warns of low pressure and indicates the UF_6 cold trap is empty when collected UF_6 is being sublimed for transfer back to a receiver cylinder (gas-back). A second low alarm, at 1 mbar (0.401 in. H_2O), closes the UF_6 cold trap outlet valve to prevent UF_6 flow to the vacuum pump. A second high alarm, at 80 mbar (32.1 in. H_2O), trips the UF_6 cold trap off-line, switching the heater/chiller unit off and closing the inlet/outlet valves.

A weighing system monitors the UF_6 contents of the UF_6 cold trap. A first alarm at 20 kg (44.1 lb) warns that the UF_6 cold trap is full. At 25 kg (55.1 lb) the UF_6 cold trap trips off-line, the inlet and outlet valves are closed, and a gas-back sequence is required.

The temperature of the UF_6 cold trap is controlled at –60°C (-76°F) during cooling to desublime any UF_6 and at 20°C (68°F) for heating during sublimation to empty the UF_6 cold trap of collected UF_6 (gas-back). A low alarm at –63°C (-81.4°F) warns of a chiller unit fault. A first high alarm at –52°C (-61.6°F) closes the UF_6 cold trap outlet valve and a second high alarm at 25°C (77°F) warns of high temperature during gas-back. At 30°C (86°F) the UF_6 cold trap trips off-line to avoid desublimation of UF_6 in the header.

D. Blending and Sampling Vent Subsystem Vacuum Pump/Chemical Trap Set.

To prevent the carbon trap from overheating and overfilling with UF_6, there are two instruments. One sensor monitors the carbon trap temperature. This sensor will close the UF_6 cold trap outlet valve when carbon trap temperature exceeds 42°C (108°F). This blocks flow to the Vacuum Pump/Chemical Trap Set. This sensor will also provide an automatic trip of the associated vacuum pump on carbon trap high temperature. The carbon trap also has a weigh system. In addition to local weight display, this system will shut down the vacuum pump when the high weight set point is reached.

The activated aluminum oxide trap on the vacuum pump/chemical trap set is also equipped with a weigh system. The weigh system on the aluminum oxide trap only displays a weight locally. There is no control function on this weight indicator.

Increase in weight is used to monitor accumulation of UF_6 in the carbon trap and HF in the aluminum oxide trap. The traps are replaced based on the accumulated weight.

3.4.6.8 Criticality Safety

3.4.6.8.1 Product Cylinders

Calculations were performed on infinite two-dimensional arrays of full 48Y or 30B product cylinders. Inside each cylinder a region of UO_2F_2/water mixture was located. The remainder of the interior of the cylinder was assumed to be filled with 6.0 $^{w}/_{o}$ ^{235}U enriched UF_6. Cylinders in the arrays were placed with the valve and base ends alternately in contact, so that the moderated region in a given cylinder was in the closest possible proximity to the moderated region in an adjacent cylinder. All cylinders were considered to be lying on a concrete pad one meter thick. Moderation was varied to obtain the optimum H/U ratio. Worst-case external reflection/moderation conditions were found by varying the density of the interstitial water between cylinders to simulate frost or snow. The calculation also assumed one cylinder above (touching) the array to simulate movement in/out/over the array.

For the 48Y cylinder, the condition that met the upper safety limit had an H/U ratio of 11.5 with an interstitial water density of 0.10 g/cm^3 (6.2 lb/ft^3). Thus, the maximum safe mass of hydrogen in each type product 48Y cylinder in an array was determined to be 1.05 kg (2.31 lb) present in the form of 9.5 kg (20.9 lb) of water.

For the 30B cylinder, the condition that met the upper safety limit had an H/U ratio of 10.5 with an interstitial water density of 0.25 g/cm^3 (15.6 lb/ft^3). Thus, the maximum safe mass of hydrogen in each type product 30B cylinder in an array was determined to be 0.95 kg (2.09 lb) present in the form of 8.5 kg (18.7 lb) of water.

Criticality safety of Type 48Y and 30B product cylinders depends on the control of moderator content. Criticality safety is achieved by ensuring that there is less than 1.05 kg (2.31 lb) of hydrogen present in a Type 48Y cylinder and less than 0.95 kg (2.09 lb) of hydrogen present in a Type 30B cylinder.

3.4.6.8.2 UF_6 Cold Trap

Although the cold trap has a large internal volume it is individually safe by shape, the trap body having an internal diameter of 20.3 cm (8.0 in). This compares with the safe diameter of 21.9 cm (8.6 in) for 6.0 $^{w}/_{o}$ enrichment. Individual cold traps are thus safe in isolation for any uranyl fluoride/water mixture. In practice the maximum H/U atom ratio in a cold trap will be 7; however, a sensitivity study is performed to determine the optimum H/U ratio, providing an additional margin of safety.

The cold trap has a minimum edge separation of 180 cm (70.9 in) from any other fixed plant vessels that can accumulate enriched uranium. The cold trap can thus be considered to be neutronically isolated from other fixed vessels.

According to the restrictions on movement of mobile vessels, one vessel can come into contact with a trap but any others have to be kept at 60 cm (23.6 in) separation.

MONK8A (SA, 2001) calculations have been performed in which a vacuum cleaner is in contact with the cold trap, and another vessel (a 14 L (3.7 gal) product vent vacuum pump) is at 60 cm (23.6 in) edge spacing from the cold trap. These are typical of Separation Plant mobile vessels. Each mobile vessel was modeled with the appropriate uranic fill; the vacuum cleaner was filled with uranyl fluoride/water mixture with optimum moderation (H/U=12), and the vacuum pump (conservatively containing hydrocarbon oil) was filled with uranic breakdown of composition $UF_4 \cdot 10.5CH_2$. The resulting k_{eff} = 0.8229 shows a slight increase in reactivity with respect to the isolated cold trap using the same conservative assumptions. The vacuum cleaner was assumed to be a cleaner of internal diameter 20.3 cm (8.0 in) and length 66 cm (26.0 in) and was assumed to be entirely filled with uranic material with an enrichment of 6.0 $^{w}/_{o}$. MONK8A (SA, 2001) calculations have been carried out for an isolated cylinder using these dimensions, filled with uranyl fluoride/water at optimum moderation and with 2.5 cm (0.984 in) water reflection. This gave a value for k_{eff} of 0.8037. The cleaner has HEPA filtration on the exhaust, and will be dedicated for cleaning operations where uranic material is involved and will be marked clearly.

Additionally, calculations were performed in which it was assumed that there are no movement controls, and both the vacuum cleaner and pump were in contact with the cold trap. Even with 2.5 cm (0.984 in) spurious water reflection placed around each unit, and at enrichment of 6.0 $^{w}/_{o}$, the result remained substantially subcritical with k_{eff} = 0.8673.

The cold trap has therefore been determined to be safe both in isolation and while interacting with other fixed plant or vessels in movement for ^{235}U enrichments up to 6.0 $^{w}/_{o}$.

3.4.6.8.3 Vacuum Pump / Chemical Trap Set

These chemical traps of the Blending and Sampling Vent Subsystem are individually safe by diameter (20.3 cm (8.0 in) compared with the safe diameter of 21.9 cm (8.6 in) calculated for 6.0 $^{w}/_{o}$ enrichment). However, calculations have been performed concerning the effect of possible neutron interaction with nearby (uranium bearing) equipment.

In the MONK8A (SA, 2001) calculations, the traps were both assumed to fill entirely with uranyl fluoride/water with no restriction on water content. This is conservative, as in practice the H/U ratio of the uranyl fluoride in the traps will have a limiting upper value of 7. Also, the space within the trap, which would normally be occupied by carbon or alumina, is modeled as being filled with uranic material. This maximizes the mass of fissile material within the traps and provides added conservatism. The pump, alumina traps, oil trap and exhaust filter are assumed to be filled with uranyl fluoride/water of unlimited water content. This is conservative, as virtually no uranium is expected in these components.

Calculations were performed to account for interaction with other vessels in movement. According to the restrictions on movement, one mobile vessel can come into contact with one of the fixed chemical absorber traps, but other mobile vessels are assumed to be at 60 cm (23.6 in) separation. The case modeled was for a vacuum cleaner (of diameter 20.3 cm (8.0 in) and length 66 cm (26.0 in)) to be brought into contact with the vacuum pump in the product vent array. One other item, a 14 L (3.7 gal) rotary vane pump, was placed at 60 cm (23.6 in) edge spacing from the vacuum cleaner. The vacuum cleaner was assumed to be a cleaner of internal diameter 20.3 cm (8.0 in) and length 66 cm (26.0 in) and was assumed to be entirely filled with uranic material with an enrichment of 6.0 $^{w}/_{o}$. MONK8A (SA, 2001) calculations have been carried out for an isolated cylinder using these dimensions, filled with uranyl fluoride/water at optimum moderation and with 2.5 cm (0.984 in) water reflection. This gave a value for k_{eff} of 0.8037. The cleaner has HEPA filtration on the exhaust, and will be dedicated for cleaning operations where uranic material is involved and will be marked clearly.

The MONK8A (SA, 2001) calculation for the worst case, where all vessels were assumed to be entirely filled with uranyl fluoride/water mixture at optimum moderation, a trap and a vacuum cleaner are in contact with the fixed pump, and the pump volume is 14 L (3.7 gal), yields a k_{eff} = 0.9328.

It should be noted that the above MONK8A (SA, 2001) model represents extreme accident conditions in terms of uranium accumulation and moderator ingress. It should also be noted that the simple MONK8A (SA, 2001) model used for the vacuum pump in all of the calculations is conservative. Since the real shape of the internal free volume is far from optimum, an explicit model of the pump is expected to result in a significant reduction in k_{eff}.

The vacuum pump/chemical trap set has been shown to be safe under normal operating conditions and credible abnormal operating conditions, for ^{235}U enrichments up to 6.0 $^{w}/_{o}$.

3.4.7 Product Liquid Sampling System

The NEF Product Liquid Sampling System uses a process essentially the same as the Claiborne Enrichment Center. The NRC staff previously reviewed the Claiborne Enrichment Center license application relative to the Product Liquid Sampling System and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion on the Product Liquid Sampling System is provided in NUREG-1491 (NRC, 1994), Section 3.6. The use of a dedicated vent system, the Blending and Sampling Vent Subsystem, rather than a mobile unit as in the Claiborne Enrichment Center, is the only appreciable difference.

3.4.7.1 Functional Description

The primary function of the Product Liquid Sampling System is to provide a means to validate the precise mean concentration of uranium-235 (^{235}U) and the purity of uranium hexafluoride (UF_6) in the product by taking homogenized liquid UF_6 samples from each product cylinder. All product cylinders are sampled prior to being released for shipment to the customer.

The sampling process is carried out with UF_6 in the liquid state. At ambient temperature, the product in the 30B cylinders is in solid form when the cylinders are placed in the autoclave. Heating the cylinders in the autoclave transposes the UF_6 from the solid phase to the liquid phase. Once in the liquid phase, the cylinder is held at temperature for a sufficient period of time to assure homogenization. After homogenizing, the autoclave is tilted to pour the liquid into the sampling manifold and then into the sample bottles.

In the liquid phase, the pressure in the product cylinders is above atmospheric. The autoclaves provide a secondary confinement barrier and protection in the event a cylinder or sampling manifold should leak.

The system is shown in Figure 3.4-13, Process Flow Diagram Product Liquid Sampling System.

3.4.7.2 Major Components

The Product Liquid Sampling System consists of only one main piece of equipment – the Product Liquid Sampling Autoclave. The Product Liquid Sampling Autoclave is shown in Figure 3.4-14, Liquid Sampling Autoclave Equipment Drawing. The autoclave consists of numerous parts that are all integrated together into one machine (the autoclave). The primary parts of each autoclave are a secondary confinement barrier pressure vessel, tilting mechanism, external cooling water coils and exterior insulation. Also included inside the pressure vessel are a cylinder support frame and rails, electric air heaters and air circulation fan, and a sampling manifold. There is a stand-alone control system and instrumentation.

All components of the autoclave are constructed of materials that have been used in existing Urenco plants for over 30 years. The autoclave pressure vessel is constructed of carbon steel to ASME specifications. The sampling manifold is constructed of Monel. The autoclave is designed to sustain seismic loading without a loss of integrity.

In normal operation, the Product Liquid Sampling System is vented during sample manifold connection and disconnection via a system that is shared with the Product Blending System.

A brief description of each major component of the Product Liquid Sampling System is provided below:

A. Cylindrical Pressure Vessel (Secondary Confinement Barrier).

For sampling, the 30B product cylinders (primary confinement barrier) are loaded into the cylindrical pressure vessel (secondary confinement barrier) that is mounted horizontally. In the event of an accidental release of product during the sampling operation, the pressure vessel provides confinement of any UF_6, UO_2F_2, and HF. The pressure vessel is designed and fabricated in accordance with the requirements of ASME Section VIII, Division1 (current version at the time of autoclave manufacture), with the exception that the pressure relief devices specified in Sections UG-125 through 137 are not be provided due to the potential for release of hazardous material to the environment through a pressure relief device. Instead, two independent and diverse automatic trips of the autoclave heaters and fan motor are provided to eliminate the heat input and preclude approaching the autoclave design pressure. This is considered to be acceptable due to the large margin between the autoclave design pressure 12 bar (174 psia) and the maximum allowable working pressure 1.8 bar (26 psia) and the fail-safe design of the two independent and diverse automatic trips of the autoclave heaters and fan motor. The pressure vessel is also tested and stamped to the requirements of ASME Section VIII, Division 1 rules and is registered with the National Board. The pressure vessel design pressure is 12 bar (174 psia) absolute and the design temperature is 160°C (320°F). One end of the pressure vessel has a welded on (stationary) dished head. On the other end is a swing out door assembly that consists of a dished head, sealing ring, gaskets, and a locking device to lock the head assembly in place after the door is closed. There are dual gaskets to provide high sealing integrity. There is also a viewing port in the door head.

B. Cylinder Support Frame and Rails.

A support frame is inside the pressure vessel. The frame is designed to contain the 30B cylinder. The support frame has rails that match the rail transporter rail design. When the cylinder is inserted in the autoclave, the frame and rails prevent the cylinder from moving when the pressure vessel is tilted. The support frame also prevents the cylinder from moving out of position during any abnormal event (such as seismic).

C. Electric Heaters and Fan.

Three electric heaters heat the inside of the autoclave. In addition to the three heaters, there is one variable speed fan that provides forced circulation of hot air over the exterior of the cylinder.

D. Sampling Manifold.

A sampling manifold is connected to the cylinder isolation valve and attached to the cylinder skirt to provide mechanical support, after the cylinder is in place. The sampling manifold is a single pipe, fabricated to provide three drain points for connection of three type 1S sample bottles to the cylinder. The total volume of the sampling manifold is such that the volume of UF_6 held in the manifold, when filled, will provide a sample of the required volume into each of the three sample bottles.

E. Cooling Coils.

The autoclave is cooled with coils mounted on the exterior of the pressure vessel. Cooling media is water supplied from the Chilled Water Distribution System.

F. Insulation.

The external surfaces of the pressure vessel are insulated for energy conservation. The insulation is non-flammable.

G. Tilting Mechanism.

The tilting mechanism raises and lowers the end of the pressure vessel with the fixed head (opposite the door end), while the other end rotates around hinge pins located under the pressure vessel. The tilting mechanism provides three positions:

- When the sample manifold is being filled, the tilting mechanism sets the incline to 30° from horizontal. At this incline, liquid UF_6 pours from the cylinder into the sampling manifold.

- For cylinder loading and unloading, the tilting mechanism sets the centerline of the pressure vessel parallel to the floor (0°).
- When the cylinder is in warm-up, homogenization, and cooling, or the manifold is being cleared, the tilt mechanism sets the autoclave at –2° from horizontal.

H. Stand Alone Control System.

The autoclave has a stand-alone control system. This system and its associated instrumentation are described in Section 3.4.7.7, Instrumentation.

I. Blending and Sampling Vent Subsystem.

Venting of the Product Liquid Sampling System is performed using the same equipment as is used for the venting of the Product Blending System. The Blending and Sampling Vent Subsystem equipment consists of a UF_6 cold trap with heater and chiller unit, and a vacuum pump/chemical trap set that includes carbon and aluminum oxide traps and a vacuum pump.

3.4.7.3 Design Description

The design bases and specifications are given in Table 3.4-10, Product Liquid Sampling System Design Basis. Applicable codes and standards are given in Table 3.4-11, Product Liquid Sampling System Codes and Standards.

There are five Liquid Product Sampling Autoclaves at the NEF.

The Product Liquid Sampling System consists of autoclaves that liquefy and homogenize the UF_6 contained in international 30B cylinders. This process is accomplished by passing hot air over the cylinders at a controlled rate.

For normal operation, a filled 30B product cylinder is loaded into an autoclave by rail from the cylinder transporter, and secured by clamps to prevent movement when the autoclave is tilted.

The sampling manifold is connected to the cylinder valve and secured to the cylinder skirt. The manifold is then connected to the Blending and Sampling Vent Subsystem. It is purged with nitrogen and pressure tested, and then evacuated and vacuum tested. With the manifold evacuated, the vent system is disconnected and the cylinder valve is opened by hand. The cylinder valve is verified as open and not blocked, and the cylinder starting pressure is verified as suitable to continue. Then the manual actuator used to close the cylinder valve is connected to allow the valve to be closed from the outside of the autoclave. The manual actuators for the sample bottle valves are also connected.

The autoclave door is then closed and locked.

The autoclave is pressurized at ambient temperature to approximately 1,200 mbar (17.4 psia) absolute pressure with nitrogen. This assures a slight pressure (above atmospheric) still exists at the end of the sampling cycle, following cooling. The positive pressure allows the autoclave to vent and ensures some gas flow to the HF monitor located in the line to the Separations Building GEVS.

The autoclave is then tilted to the –2° position to reduce the potential for splash over of UF_6 into the manifold during heat-up. The electric heaters and fan are then actuated and the internal temperature in the autoclave is brought up to operating temperature.

Hot air forced over the cylinder raises the UF_6 temperature to change the solid UF_6 to liquid. When the measured UF_6 pressure reaches its control set point and the cylinder contents are in equilibrium, the temperature set point remains steady.

When the pressure set point of 2.5 bar (36.3 psia) is reached, the autoclave maintains the pressure and temperature so the UF_6 can homogenize. This homogenizing period lasts for approximately 16 hours.

After homogenization, the sampling procedure begins. With the sample bottles closed, the heater controller is changed over to temperature control and the set point for the air temperature is elevated slightly. Due to the much smaller mass of the sample manifold compared to the cylinder, the sample manifold will heat up quicker than the cylinder. Any liquid UF_6 within the sample manifold piping vaporizes and flows back into the cylinder and condenses.

The air heaters and fan are then switched off.

After the heaters and fan are off, the autoclave is tilted to 30°. The liquid UF_6 flows from the product cylinder into the sampling manifold (which has three 1S sample bottles connected to it). To avoid overfilling of the bottles, the volume of the pipe on each branch from the manifold to the bottle is less than the volume of the sample bottle.

After pouring liquid UF_6 into the sampling manifold, the autoclave is returned to the –2° position and the valves on the sample bottles are opened to fill the bottles with liquid UF_6. The valves of the sample bottles are then closed.

The air heaters and fan are switched on and the temperature set point is increased slightly. The remaining liquid UF_6 within the sampling manifold is vaporized and re-condenses in the cylinder. This removes any residual liquid UF_6 from the manifold.

Following the sampling operation and removal of the residual liquid UF_6 from the manifold, the cylinder valve is closed. The autoclave and the cylinder are cooled down by circulating cooling water through the cooling coils until the pressure in the cylinder is subatmospheric and the liquid UF_6 goes back to the solid state.

The autoclave is then returned to the horizontal position. Once the autoclave is validated to be free of any UF_6 and HF, the door is opened.

The sample manifold is purged with nitrogen and vented to the Blending and Sampling Vent Subsystem UF_6 cold trap and vacuum pump/chemical trap set.

The three sample bottles are removed and taken to the laboratory. One bottle is analyzed, one is sent to the customer, and one is held as a reference sample.

The cylinder is then removed from the autoclave by the cylinder transporter.

3.4.7.4 Interfaces

The Product Liquid Sampling System interfaces with the following systems and utilities.

A. Blending and Sampling Vent Subsystem

B. Separations Building GEVS

C. Chilled Water Distribution System

D. Nitrogen System

E. Compressed Air System

F. Electrical System

G. Hoisting and Transportation Equipment

H. Plant Control System.

3.4.7.5 Design and Safety Features

The Product Liquid Sampling System is designed and constructed to provide safe operation for plant personnel as well as the general public. Releases to the atmosphere are minimized by:

A. Cylinder fill mass is limited to ensure cylinder integrity by verifying the weight of product cylinder is within limits before placement and heating in the autoclave.

B. Any heating, handling, or sampling of UF_6 in its liquid state is done in a sealed autoclave to provide secondary confinement in the event of leakage of the primary confinement barrier. The autoclave is not opened until the UF_6 is cooled to a solid and the cylinder is returned to less than atmospheric pressure.

C. Temperature in each autoclave, and of the cylinder being sampled, is monitored and controlled.

D. Abnormal temperature in each autoclave is detected via temperature sensors and indicated by alarms. Appropriate actions to shut down the systems are taken as necessary.

E. Abnormal pressure in each autoclave, and in the cylinder being sampled, is detected via pressure sensors and indicated by alarms. Appropriate actions to isolate the process or shut down the systems are taken automatically.

F. Before opening the autoclave or disconnecting the sampling manifold, the equipment and process piping is evacuated and purged with nitrogen.

G. A local exhaust to the Separations Building GEVS is provided any time the autoclave is opened or the sample manifold is disconnected.

H. Before discharge to the Separations Building GEVS, the vent gases flow through the UF_6 cold trap and then across activated carbon and aluminum oxide in the Blending and Sampling Vent Subsystem to remove any traces of UF_6 and HF.

I. Temperature and weight in the Blending and Sampling Vent Subsystem carbon trap is monitored and a trip on weight and a trip on temperature stops the Blending and Sampling Vent vacuum pump.

J. The autoclave is designed and tested to ensure leak tight integrity is maintained.

K. The autoclave door seal is leak tested and inspected prior to each autoclave sample sequence.

3.4.7.6 Operating Limits

The Product Liquid Sampling System is capable of handling the sampling requirements of the entire plant. The system is designed to allow flexibility by providing for the sampling of up to an equivalent of nine product cylinders per week. This number provides a margin based on the 3,000,000 SWU per year rated capacity of the NEF.

3.4.7.7 Instrumentation

Each autoclave is controlled by a stand-alone control system. This system carries out all the control and protection functions as well as providing interface with the Plant Control System. There is a local operator interface (LOI) at each autoclave. From the LOI an operator can control all functions of the autoclave, as well as start and stop the autoclave process. All process variables are displayed at the LOI and are relayed to, and displayed in, the Control Room.

The process variables, such as pressures, temperatures, and interlock positions, are automatically controlled. Deviations from specified values are detected and indicated via two level alarm systems. At the first alarm level, the process operator has the ability to manipulate the process to restore it to normal. At the second alarm level, automatic action is taken to provide system protection. For safety, system protection, and operability, some critical sensors are duplicated. Action is initiated if any one out of the two sensors reach alarm levels.

A. Product Liquid Sampling Autoclave.

Two pressure sensors, connected to the cylinder by the sampling manifold, monitor and control the cylinder pressure during heating, homogenization, and sampling. Normal pressure during homogenization and liquid sampling is less than 3.0 bar (43.5 psia). The first alarm level is 3.0 bar (43.5 psia) to give operator warning of over pressurization. The second alarm level is 3.2 bar (46.4 psia), which automatically de-energizes the air heater and fan. A second cylinder pressure monitor with the same alarm levels provides backup protection.

Pressure inside the autoclave is monitored with a single sensor. A first high switch, at 1.1 bar (16 psia), prevents the door from being opened while the autoclave is under pressure. A second high switch at 1.2 bar (17.4 psia), which is the normal operating pressure of the autoclave at the start of heating, closes the nitrogen supply valve. The third high alarm level, at 1.5 bar (21.8 psia), gives the operator warning of over pressurization. The final high alarm level is 1.8 bar (26.1 psia) and automatically de-energizes the autoclave heaters and aborts the cycle – manual resetting of the sample cycle is required.

A temperature sensor monitors the surface of the cylinder during heating and cooling. A temperature above 55°C (131°F) prevents the autoclave door from being opened. This ensures that the UF_6 is solid before the cylinder can be removed from the autoclave.

Dual temperature sensors monitor the autoclave air temperature for control and protection. One sensor modulates power to the heaters to control the autoclave air temperature. The other sensor provides no control, but monitors and protects the autoclave air temperature only. Both sensors provide protection by a one from two voting system. Normal temperature during heating is less than 110°C (230°F). A first switch at 40°C (104°F) prohibits unlocking the autoclave door until the autoclave has cooled at the end of the sampling cycle. An alarm at

110°C (230°F) warns the operator of high temperature. The third alarm level at 115°C (239°F) automatically de-energizes the autoclave heater and fan.

Each of the three autoclave heater elements has a temperature switch at 150°C (302°F) to protect the element. The air circulating fan motor is protected using a temperature sensor with a high warning alarm and a switch to de-energize the heaters and fan.

The air quality of each autoclave is monitored for the presence of HF. If HF is detected, indicating a breach in the primary containment (cylinder or sampling manifold), the autoclave vent valve and the door are prevented from opening. A second HF monitor in the common vent header from the autoclaves to the Separations Building GEVS provides a backup check to verify the quality of the air venting from the autoclave. If HF is detected here, an alarm signals to manually close the autoclave vent valve, and the other autoclave vent valves cannot be opened.

In addition to the process control noted above, there are six timers associated with the various steps of the sampling cycle.

Two timers provide for monitoring the autoclave to maintain safe start-up of the heating cycle. The value of these two timers is made to enable monitoring of the autoclave pressure rise during the start of the heating cycle verses time. The autoclave pressure is compared to an algorithm during the first phase of the heating stage when the heating is carried out with a preset air temperature. If the pressure rise conforms to the algorithm, the heating is permitted to advance to a second phase where the heating is controlled by the cylinder pressure. In the event the algorithm is not being met, the heating cycle is aborted.

Two other timers operate to monitor the quality of the air space in the autoclave and support the operation of the internal HF monitor. After the system stabilizes, the autoclave air pressure and temperature are compared. A departure from the anticipated pressure to temperature ratio indicates a leak has occurred. A lower than anticipated pressure to temperature ratio indicates a pressure leak from the secondary containment (autoclave). A higher than anticipated ratio indicates a leakage of UF_6 into the secondary containment. If the pressure/temperature ratio is outside the anticipated range, the cycle is aborted.

Another timer is used to confirm that the cooling cycle is continued for a sufficient time to ensure the cylinder contents are solidified before the cylinder is removed from the autoclave.

A final timer ensures that the autoclave is fully vented before the autoclave door is opened.

B. Blending and Sampling Vent Subsystem.

The instrumentation for the Blending and Sampling Vent Subsystem equipment is discussed in Section 3.4.6, Product Blending System.

3.4.8 Contingency Dump System

The NEF Contingency Dump System uses a similar process to the original Claiborne Enrichment Center. The NRC staff previously reviewed the Claiborne Enrichment Center SAR application relative to the Contingency Dump System and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion on the Contingency Dump System is provided in NUREG-1491 (NRC, 1994), Section 3.5. The primary differences are:

A. The number of chemical traps has been increased to three per cascade.

B. Pumping systems supporting the traps have been dedicated to single cascades rather than per assay unit.

These changes reflect the increased cascade size.

3.4.8.1 Functional Description

The Contingency Dump System provides an exhaust route for UF_6 from the cascade in the event of the cascade operating outside of its design envelope. The Contingency Dump System also provides an evacuation route for UF_6 and light gases to allow the centrifuges to be safely run down to rest.

The Contingency Dump System is shown in Figure 3.4-15, Process Flow Diagram Contingency Dump System.

The Contingency Dump System forms only part of the dumping philosophy. Dumping of the UF_6 from the cascade, should the need arise, will take place by first choice to the Tails Take-off System. If the Tails Take-off System becomes unavailable, the Contingency Dump System is used. The Contingency Dump System is designed to operate in one of two principal operating modes, passive evacuation or active evacuation. The function of the passive evacuation mode is to trap the UF_6 evacuated from the cascade in the sodium fluoride (NaF) traps. This "passive evacuation" is so called because evacuation of the cascade can initially take place without actively pumping; the low pressure maintained in the NaF traps and buffer volume in standby mode facilitates this process. Operation in the passive evacuation mode results in a progressive increase in the operating pressure at the NaF traps due to the accumulation of light gas in the buffer volume. This light gas is removed from the buffer volume by operation in the active evacuation mode. In "active evacuation" the buffer volume is opened to the vacuum pump/chemical trap set and the light gas is exhausted from the passive system via the carbon and aluminum oxide traps to the Separations Building GEVS.

3.4.8.2 Major Components

The major components of the Contingency Dump System are listed below.

A. Contingency Dump System NaF Traps and Buffer Volume.

A pressure transducer is located on the cascade header to monitor conditions at the cascade header during dump. This transducer is dedicated to the Dump Control System and provides an indication of cascade conditions during dump.

The Contingency Dump System uses three chemical traps filled with sodium fluoride (NaF). An NaF trap is shown in Figure 3.4-16, NaF Trap Equipment Drawing. This material is able to adsorb UF_6 and HF without producing gaseous reaction products. The buffer volume provided after the NaF traps accommodates any light gas that passes through the NaF traps. The NaF traps and buffer volume constitute the "passive" part of the Contingency Dump System. This passive part of the Contingency Dump System is able to maintain a dump capacity in the event of a loss of other services or utilities.

Manual valves are fitted to the inlet and the outlet of the NaF traps and buffer volume to act as protective barriers during maintenance activities. Automatic valves are provided for plant operation. Pressure transducers are positioned in the Contingency Dump System to monitor both the buffer volume pressure and dump pump suction pressure. This monitoring is for both the operation and protection of the Contingency Dump System and the prevention of backflow of light gases through the NaF traps to the Cascade System.

A fourth pressure transducer is mounted at the cascade valve frame between the automatic and manual valve to enable monitoring of the seating efficiency of these two valves. A tight shut-off of the valve must be maintained throughout the life of the Contingency Dump System to prevent the NaF traps becoming loaded with UF_6. A tight shut-off valve is required to enable maintenance of the Contingency Dump System.

B. Contingency Dump System Vacuum Pump/Chemical Trap Set.

The major components of the Contingency Dump System Vacuum Pump/Chemical Trap Set are:

- A roots type and rotary vane vacuum pump
- Activated carbon trap
- Aluminum oxide trap.

The NaF traps and buffer volume of the passive dump system are backed by the Contingency Dump System Vacuum Pump/Chemical Trap Set which comprises, in order, a Roots type vacuum pump, activated carbon trap, aluminum oxide traps and sliding vane type vacuum pump. The sliding vane vacuum pump discharges through a final oil trap into the Separations Building GEVS. Connection of the Contingency Dump System vacuum pump/chemical trap set is made to the NaF traps/buffer volume of the Contingency Dump System by flexible stainless steel vacuum bellows and to the Separations Building GEVS by a pressure hose. The equipment is assembled as a modular package to facilitate easy replacement and maintenance of the unit as a whole in the event of a failure.

The function of the activated carbon trap is to remove small traces of UF_6 and the aluminum oxide trap is to remove any HF from the gas flow. These traps are fitted upstream of the sliding vane vacuum pump. A second, smaller, aluminum oxide trap, is fitted immediately before the sliding vane vacuum pump. This trap prevents back diffusion of oil from the vacuum pump into the traps. The pump discharge trap prevents oil entering the Separations Building GEVS.

In order to measure any accumulated mass within the activated carbon trap and aluminum oxide trap a local facility for weighing each trap without disturbing the process is provided.

To maintain a high availability of the Contingency Dump System, power supply to the Contingency Dump System pumps is maintained by standby diesel generators in the event of a failure of the normal power supply. Each cascade has one Contingency Dump System with no installed redundancy.

3.4.8.3 Design Description

The design bases and specifications are given in Table 3.4-12, Contingency Dump System Design Basis. Applicable codes and standards are given in Table 3.4-13, Contingency Dump System Codes and Standards.

An independent Contingency Dump System is provided for each cascade. All components of the Contingency Dump System operate a subatmospheric pressure. Release of UF_6 or light gases are minimized because leakage, if it were to occur, would be inward to the system.

All of the process equipment in the Contingency Dump System is designed, constructed, and operated using good engineering practice and in accordance with the LES Quality Assurance program.

The materials of construction, corrosion allowances and fabrication specifications for the equipment and piping used in the Contingency Dump System are compatible with UF_6 and HF at the operating conditions and have been proven by extensive use in existing enrichment plants.

3.4.8.4 Interfaces

The Contingency Dump System interfaces with the following systems and utilities:

A. Cascade System

B. Separations Building GEVS

C. Nitrogen System

D. Compressed Air System

E. Electrical System

F. Plant Control System.

3.4.8.5 Design and Safety Features

This system is designed and constructed to provide safe operation for plant personnel as well as the general public. Principal design features are as follows:

A. All piping, vessels and pumps in the Contingency Dump System operate at subatmospheric UF_6 pressure.

B. Piping is all welded construction and process valves are bellow sealed.

C. Before carrying out any disconnections or connections of equipment, the piping is evacuated and nitrogen purged. Flexible exhaust hoses connected to the Separations Building GEVS remove any releases from the work area.

D. Before discharge to the Separations Building GEVS, all gases flow across activated carbon and aluminum oxide to remove any traces of UF_6 and HF via the Contingency Dump System Vacuum Pump/Chemical Trap Set.

E. Monitoring of fill level of NaF trap when charging the NaF trap.

F. Hydrocarbon lubricants are not used. The rotary vane vacuum pumps are lubricated with fully fluorinated synthetic oil such as "Fomblin," a perfluorinated polyether (PFPE).

G. The potential for capture of UF_6 and HF in the NaF traps is maximized by operation of the Contingency Dump System in a passive mode. In passive evacuation mode the flow of UF_6 from the cascade is restricted to the NaF traps and buffer volume by valving.

H. The main electrical supply is supported by a Standby Diesel Generator System for electrical services essential to equipment protection. In the case of a power failure the UF_6 valves will retain their position because their control is via a 24 VDC uninterruptible power supply (UPS). On loss of the UPS the valves will revert to a fail-safe position.

I. Compressed air has a high reliability in normal operation with sufficient capacity at the pressure reservoir for a safe shut down. To protect against a compressed air failure, all air driven valves are fitted with check valves to ensure that the valve retains a position of at least 50% for six hours.

J. The potential for a criticality arising at the Contingency Dump System is eliminated by ensuring a safe design. Both the NaF traps and the buffer volume are designed and installed to be geometrically safe.

K. Weight in the contingency dump vent vacuum dump/chemical trap set carbon trap is monitored and a trip on weight stops the contingency dump vent vacuum pump.

3.4.8.6 Operating Limits

The Contingency Dump System must be able to remove the UF_6 content of the cascade and evacuate to a minimum pressure during abnormal operating conditions.

3.4.8.7 Instrumentation

The cascade protection system is provided by two Programmable Logic Controllers (PLCs), one PLC controlling and protecting the process while the other PLC monitors parameters essential to the separation process and takes action if these parameters are out of specification. In the event of a failure of either of the PLCs, the failure will invoke a cascade dump.

The Contingency Dump System process variables such as pressures and valve positions are displayed in the Control Room and are automatically controlled by the Contingency Dump System Local Control Center (LCC). Deviations from the specified values are detected and indicated via two-tiers of signals. At the first level the signal provides an alarm only and the process operator has the ability to manipulate the process to restore it to normal operation. At the second alarm/trip level, automatic action is taken to provide system protection.

The pressure transducers and valve and pump status signals of the Contingency Dump System are directly connected to the control PLC in the Contingency Dump System LCC.

The dump system has two distinct modes of operation, in the normal state the Contingency Dump System is in standby mode. In the event of a "dump" signal the "dump" mode control and action set-points will override the trips and alarms of the standby mode where these set-points are different.

The system is placed in Dump Mode either automatically by a dump demand signal from the cascade control and protection system or can be manually selected either by a push button in the Control Room (Cascade Hall Dump) or from the Plant Control System (Cascade Dump).

A. Contingency Dump System NaF Traps and Buffer Volume.

The NaF traps and buffer volume comprise the passive part of the Contingency Dump System.

The Contingency Dump System pressure is monitored at two positions at the traps and buffer volume. The first position is at the buffer volume upstream of the automatic shut off valve. The second position is downstream of the shut-off valve and monitors the vacuum pump suction line pressure.

The passive dump system operating pressure at the NaF traps and buffer volume is maintained within the range high (H1) to low (L) while the system is in the Standby mode.

Pressure control maintains the pressure at the NaF traps and buffer volume by opening the downstream valve on rising pressure (H1) and closing the valve on falling pressure (L).

A high alarm (H2) at the NaF traps indicates an alarm in the event of the buffer volume pressure rising above its normal operating range in standby mode. A high-high alarm (HH) inhibits the use of the Contingency Dump System by removing the "dump system available" signal to the cascade protection system.

Pressure indication downstream of the automatic valve provides a safety and monitoring function. In the event of a high-high pressure an alarm/trip (HH2) inhibits the use of the active evacuation sequence and will close the valve. The HH2 alarm/trip is active during all standby and dump operating modes of the Contingency Dump System. In "dump" mode the HH2 alarm/trip is overridden in "light gas evacuation" mode only by alarm/trip HH1. Operation of the HH1 alarm/trip will close the valves downstream of the buffer vessel and the active evacuation valve. The low set point of the HH1 trip provides a more rapid response to a fault condition and air ingress at the lower operating pressures of the Contingency Dump System when in light gas evacuation mode.

On dump instruction the Contingency Dump System status is promoted from "Standby" to "Passive Evacuation" and UF_6 and light gas enters the Contingency Dump System from the cascade under the control of the Contingency Dump System. The buffer volume pressure indicator/controller high trip, (H3), is made active overriding the lower trip points to permit light gas passing the NaF traps to fill the buffer volume.

The time T1 is started on dump demand. Time T1 retains the Contingency Dump System in "passive evacuation" for the set period.

B. Contingency Dump System Vacuum Pump/Chemical Trap Set.

On timeout of the timer T1 or a low pressure trip at the cascade header pressure the dump sequence is promoted to "Active Evacuation," the valve down stream of the buffer volume is opened and time T2 is started. During "Active Evacuation" the Contingency Dump System pump module is used to evacuate the accumulating light gases from the buffer volume via the downstream valve. On timeout of timer T2 the Contingency Dump System enters "Light Gas Evacuation" and the cascade is evacuated through the NaF trap bypass line.

A temperature alarm is fitted to the activated carbon trap to provide indication of an excessive carry over of UF_6 gas from the NaF traps and buffer volume when in "Active Evacuation" or directly from the cascade when operating in "Light Gas Evacuation." The temperature alarm provides an alarm function only on excessive UF_6 gas flow at the activated carbon trap. The carbon trap also has a weigh system. In addition to local weight display, this system will shut down the vacuum pump when the high weight set point is reached.

The Contingency Dump System interfaces with the Cascade System to provide the Control Room operator with cascade data in the event of a failure in the cascade control PLC.

The following cascade status conditions are monitored by the Contingency Dump System PLC:

A. The position of the cascade dump valve (open/closed)

B. Recipient temperature

C. Cascade header pressure.

The Contingency Dump System monitors the pressure of the cascade header by a single pressure transducer. This pressure transducer is used in conjunction with pressure control at the Contingency Dump System buffer volume to determine the availability of the Contingency Dump System. Contingency Dump System availability is maximized over the whole of the cascade run-down by a two stage monitoring of the cascade header pressure.

Due to the anticipated infrequent use of the Contingency Dump System, its availability is maintained by a regular testing program of both monitoring equipment and valves to ensure that a failure of the Contingency Dump System PLC is revealed.

3.4.8.8 Criticality Safety

The average enrichment of the UF_6 being dumped from a cascade depends on the product and tails enrichments. Within the ranges of product enrichment up to 5.0 $^{w}/_{o}$ ^{235}U and tails depletion to 0.34 $^{w}/_{o}$ ^{235}U, the average enrichment of the UF_6 being dumped is always less than 1.5 $^{w}/_{o}$ ^{235}U. Based on this, the contingency dump traps will be analyzed at an enrichment of 1.5 $^{w}/_{o}$ rather than 6.0 $^{w}/_{o}$. The contingency dump traps are sodium fluoride traps with an inside diameter of approximately 54 cm (21.3 in).

MONK8A (SA, 2001) calculations have been carried out first for an isolated trap with 2.5 cm (0.984 in) of water reflection around the trap body. The model assumed that adsorbed UF_6 within the trap is converted to $UO_2F_2 \cdot 3.5H_2O$, i.e., the accident condition with air inleakage. The uranium enrichment was 1.5 $^{w}/_{o}$ ^{235}U. The value of k_{eff} obtained was 0.6466. The model represents a UF_6 loading in the trap of approximately 220 kg (485 lb), which would require many dumps to achieve. Contingency dump traps are thus intrinsically safe by a very large margin.

Considering interaction between the three closely spaced traps, criticality safety is demonstrated by comparison with the MONK8A (SA, 2001) calculations for storage of contingency dump traps in unspaced linear arrays. The calculation modeled a linear array of seven touching dump traps with three other vessels at 60 cm (23.6 in) spacing from the array (a residue container, a vacuum cleaner cylinder and a UF_6 pump unit). An additional dump trap was also placed in contact with the center trap of the linear array. The value of k_{eff} obtained was 0.8537. The modeled arrangement is more conservative than three spaced traps interacting with the same mobile vessels and it can be concluded that contingency dump traps are safe when interacting with any mobile vessels that are likely to be present. The vacuum cleaner was assumed to be a cleaner of internal diameter 20.3 cm (8.0 in) and length 66 cm (26.0 in) and was assumed to be entirely filled with uranic material with an enrichment of 6.0 $^{w}/_{o}$. MONK8A (SA, 2001) calculations have been carried out for an isolated cylinder using these dimensions, filled with uranyl fluoride/water at optimum moderation and with 2.5 cm (0.984 in) water reflection. This gave a value for k_{eff} of 0.8037. The cleaner has HEPA filtration on the exhaust, and will be dedicated for cleaning operations where uranic material is involved and will be marked clearly.

3.4.9 Gaseous Effluent Vent Systems

The function of the GEVS is to remove particulates containing uranium, and HF from potentially contaminated process gas streams. Prefilters and absolute filters (HEPA) remove particulates and potassium carbonate impregnated activated carbon filters are used for the removal of any HF. Electrostatic filters remove oil vapor from the gaseous effluent associated with exhaust from vacuum pump/chemical trap set outlets wherever necessary.

The systems produce solid wastes from the periodic replacement of prefilters, absolute filters, and chemical filters. The systems produce no gaseous effluents of their own, but discharge effluents from other systems after treatment to remove hazardous materials.
There are two GEVSs for the plant. The Separations Building GEVS and the TSB GEVS. Applicable codes and standards are given in Table 3.4-14, Gaseous Effluent Vent System Codes and Standards.

3.4.9.1 Separations Building Gaseous Effluent Vent System

The GEVS for the Separations Building provides exhaust of potentially hazardous contaminants. The system is shown on Figure 3.4-17, Process Flow Diagram Gaseous Effluent Vent System Separations Building, Sheets 1 and 2.

The GEVS system serving the Separations Building is located in the TSB on the first floor. The system is operated from the Control Room.

3.4.9.1.1 Functional Description

The Separations Building GEVS interfaces with the following systems, auxiliary activities, and utilities:

A. UF_6 Feed System

B. Product Take-off System

C. Tails Take-off System

D. Product Blending System

E. Product Liquid Sampling System

F. Contingency Dump System

G. Compressed Air System

H. Electrical System

I. Control Room

The design requirements provide a large safety margin between normal and accident conditions so that no single failure could result in the release of significant hazardous material. The amounts of UF_6 in the system also preclude the release of significant quantities of hazardous material from a single failure or multiple failures. Instrumentation is provided to detect abnormal process conditions so that the process can be returned to normal by operator actions.

3.4.9.1.2 Major Components

The Separation Building GEVS consists of the following major components.

A. Duct system

B. Electrostatic filter

C. Prefilters

D. High Efficiency Particulate Air (HEPA) Filters

E. Activated carbon filters

F. Centrifugal Fans

G. Monitoring and controls (HF) before and after filters

H. Automatically controlled inlet and outlet isolation dampers

I. Exhaust stack

J. Gamma monitors and controls (prefilters, HEPA Filters, and electrostatic precipitator)

K. Monitoring and controls (alpha and HF) in exhaust stack

L. Stack sampling system.

3.4.9.1.3 Design Description

The design bases and specifications are given in Table 3.4-15, GEVS Design Bases (Separations Building).

One Separation Building GEVS serves the entire Separations Building. It consists of a duct network that serves all of the uranium processing systems and operates at negative pressure. It is sized to handle the flow from all permanently ducted process locations, as well as up to 13 flexible exhaust hose exhaust points at one time. The flexible exhaust hoses are used for cylinder connection/disconnection or maintenance procedures. A minimum velocity of 12.7 m/s (2500 ft/min) is maintained in the duct system in order to ensure that particulate contaminants are conveyed through the ductwork without settling. Each section of the duct system has an orifice plate to maintain a minimum air velocity. Each section also has a damper to balance the individual flows in the system. The flexible exhaust hoses will have a capture velocity of 0.75 m/s (148 ft/min).

The ductwork is connected to two parallel filter stations. Each is capable of handling 100% of the effluent. One is online and the other is a standby. Each station consists of an 85% efficient prefilter, a 99.97% efficient HEPA filter, and a 99% efficient activated carbon filter for removal of HF. Electrostatic filters have an efficiency of 97%. The filter stations vent through one of two fans. Each fan is capable of handling 100% of the effluent. One fan is online, and the other is a standby. A switch between the operational and standby systems can be made using automatically controlled dampers. The system capacity is estimated to be 11,000 m^3/hr (6,474 cfm). A differential pressure controller controls the fan speed and maintains negative pressure upstream of the filter station. Flow rates and capacity are preliminary and are subject to change during final design.

Gases from the UF_6 processing systems pass through the prefilter which removes dust and protects the HEPA filter, then through the HEPA filter which removes uranium aerosols (mainly UO_2F_2 particles), then through the potassium carbonate impregnated activated carbon filters which captures HF. The remaining clean gases pass through the fan, which maintains the negative pressure upstream of the filter stations. Finally, the clean gases are discharged through a roof top exhaust stack on the TSB. One exhaust stack is common to the operational system and the standby system.

The materials of construction, corrosion allowances, and fabrication specifications for the equipment and ductwork used in the GEVS are compatible with UF_6 and HF and are noncombustible.

The Separations Building GEVS provides the ventilation and hazardous contaminant removal for the following systems, equipment, and areas.

It is connected via permanently ducted locations to:

A. The UF_6 Feed System, The Product Take-off System, the Tails Take-off System, the Product Blending and Sampling Vent Subsystem and Contingency Dump System.

B. All Liquid Sampling System autoclaves.

C. All discharge lines from mobile vacuum pump sets.

It is connected via flexible exhaust hoses to places where piping is normally disconnected or equipment is opened, such as:

A. The Product Take-off System and Tails Take-off System pumping trains and the UF_6 Feed Purification Subsystem, Product Vent Subsystem, Tails Evacuation Subsystem and Product Blending and Sampling Vent Subsystem vacuum pump/ chemical trap sets.

B. The Liquid Sampling System autoclaves. The lines for the flexible duct are run to a point within approximately 0.9 m (3 ft) of each door opening. Approximately 1.8 m (6 ft) of flexible duct is connected to this point to enable access to all places where the autoclave UF_6 pipework is connected/disconnected.

C. The Product and Tails Low Temperature Take-off Stations.

D. The Solid Feed Stations and Feed Purification Low Temperature Take-off Stations.

E. The Blending Donor Stations and Blending Receiver Stations.

If the Separations Building GEVS stops operating, material within the duct will not be released into the building because each of the Separations Building GEVS connections has a P-trap to catch entrained material that could otherwise fall back into the building from the ductwork during system failure.

Mobile vacuum pump units that vent to the Separations Building GEVS are available in the UF_6 Handling Areas and the Product Blending and Liquid Sampling Area.

3.4.9.1.4 Design and Safety Features

The Separations Building GEVS is designed to protect plant personnel against uranium and HF exposure. Potential hazards include the release of UF_6 and HF to the building and/or environment, contaminated filters, and contaminated oil.

The system filters contaminated gases, and continuously monitoring exhaust gas flow to the atmosphere. HF monitors and alarms are installed upstream of the filtration systems and immediately upstream of the exhaust stack to avoid the release of hazardous materials to the environment. A fault alarm is generated, in the event of a fault occurring within any of the monitors. The alarms are monitored in the Control Room.

Gamma monitors measure the build up of ^{235}U on prefilters, HEPA filters and on the electrostatic filter. Upon detection of high-high gamma levels in the Separations Building GEVS filter, the operating Separations Building GEVS train trips. Upon detection of high-high gamma levels in the Separations Building GEVS electrostatic precipitator, the trip realigns dampers to bypass and isolate the electrostatic precipitator.

The Separations Building GEVS unit is located in a dedicated room with the GEVS from the TSB. The filters are bag-in/bag-out. The frequency of filter replacement will be determined during the design phase and this section will be revised accordingly.

The Separations Building GEVS provides for continuous monitoring and periodic sampling of the gaseous effluent in the exhaust stack in accordance with the guidance in Regulatory Guide 4.16 (NRC, 1985).

The Separations Building GEVS is designed to meet all applicable NRC requirements for public and plant personnel safety and effluent control and monitoring. The system designs also comply with applicable standards of OSHA, EPA, and state and local agencies.

The design and in-place testing of the Separations Building GEVS will be consistent with the applicable guidance in Regulatory Guide 1.140 (NRC, 2001), ASME AG-1-1997 (ASME, 1997), and ASME N510-1989 (ASME, 1989). The system includes potassium carbonate impregnated activated charcoal filters for HF removal. As such, the portions of Regulatory Guide 1.140 (NRC, 2001), ASME AG-1-1997 (ASME, 1997), and ASME N510-1989 (ASME, 1989), which address activated charcoal filters for radioiodine removal are not applicable. The prefilter efficiency (85%) is based on testing in accordance with ASME AG-1-1997 (ASME, 1997). The HEPA filter efficiency (99.97%) is based on removal of 0.3 micron particles when tested in accordance with ASME-AG-1 (ASME, 1997). The impregnated charcoal filter efficiency (99%) for removal of HF is based on Urenco specifications. In-place testing and inspections of the filters will be performed in accordance with the guidance in Regulatory Guidance 1.140 (NRC, 2001). The frequency for performance of in-place filter testing and the acceptance criteria for penetration and leakage (or bypass) will be consistent with the guidance in Regulatory Guide 1.140 (NRC, 2001). Qualification testing, to verify HF removal efficiency, of the impregnated charcoal will be performed using ASTM D6646-03 (ASTM, 2003), modified to reflect removal of HF instead of hydrogen sulfide. Laboratory testing of the impregnated charcoal filter of charcoal samples will be performed on an annual basis. Throughout the useful life of the impregnated charcoal, the impregnate is progressively consumed. The laboratory testing will determine the impregnant content within the sample. The amount of impregnant present in the sample is indicative of the remaining life of charcoal bed for removal of HF.

3.4.9.1.5 Instrumentation

The process variables, pressure, fan speed, and damper positioning are all controlled automatically. The fan speed is automatically controlled to maintain negative pressure in the system. HF monitors measure the concentration of the gas in the air stream. Also, devices are used to measure the level of radiological contamination (alpha only) present in the air stream located in the exhaust stack. Deviations from specified values are indicated by alarms. HF monitors and alarms are installed upstream of the filtration system and immediately upstream of the exhaust stack to avoid the release of hazardous materials. The HF and radiological monitoring devices have non-interruptible power supplies in order to continue to function during a general power failure.

HF monitors and alarms are installed upstream of the filtration systems and immediately upstream of the exhaust stack to prevent the release of hazardous materials.

The differential pressure across the prefilter and HEPA filter is monitored to indicate required filter changes.

The GEVS control system is mounted in a Local Control Center (LCC). This is a stand-alone system that does not generate alarms during normal operation. The LCC provides automatic control of the fans and dampers and provides local control via a Local Operator Interface (LOI) that is mounted in the LCC.

The Central Control System (CCS) has no supervisory control over the Separations Building GEVS control system. However, the Separations Building GEVS LCC communicates with the CCS via the dual redundant process network so that comprehensive monitoring of the GEVS status exists. Data that is monitored is fans status, filter and duct pressure measurements, damper status, and electrostatic precipitator status. System alarms are relayed to the CCS.

The Separations Building GEVS LCC has one PLC that provides all automatic control and protection required for the system, and also the communication interface to the PCS. All equipment related to the Separations Building GEVS is directly wired to the LCC.

The radiological activity and HF monitoring instruments are stand-alone and powered separately. These instruments interface with the Separations Building GEVS LCC via hardwired signals that indicate when alarm limits have been exceeded. These alarms are overridden during calibration.

3.4.9.1.6 Criticality Safety

There are two sources of uranic material to the Separations Building GEVS, flexible exhaust hoses and rotary pump exhausts.

The rotary pump exhaust gas arising from the Product Vent Subsystem passes from the UF_6 cold trap through the activated carbon trap and alumina trap and finally through the rotary pump.

Excessive carry over from the cold trap to the carbon trap is avoided by the closure of a valve in the interconnection by a low pressure or a high temperature trip in the cold trap. The exhaust gas then passes through a trap filled with carbon that reacts irreversibly with the UF_6 and then passes through an activated alumina to remove HF. The gas is then pumped out into the Separations Building GEVS for final clean up. These chemical traps are replaced at regular intervals or when the weight indicators show that there is significant build up of material. A weight trip on the carbon trap isolates the process line from the Separations Building GEVS when the traps are about to become saturated.

The flexible exhaust hoses will be used to support product (and feed and tails) cylinder and pump changeout and maintenance activities in the separations plant and trace enriched particulate matter may be released.

The potentially oil bearing inflow to the Separations Building GEVS from the rotary vacuum pumps exhausts is first passed through an electrostatic precipitator to remove the aerosol oil before joining the rest of the effluent gas. It then passes through pre filters, HEPA filters for particulates removal and impregnated carbon filters for removing HF. Prior to the HEPA filters there is a fluoride monitor that will alarm if the concentration of the fluorine compounds within the air being drawn into the filters exceeds a pre-determined level. This will provide assurance that accumulation of uranium in the filters is not occurring. The filters are equipped with differential pressure indicators and ^{235}U selective gamma monitors that will trip on blockage or build-up of material. The amount of uranium in the electrostatic precipitator will also be monitored for gamma radiation to ensure that any slow, chronic accumulation of fissile material does not pose a hazard.

The carbon trap weight trip and Separations Building GEVS filter gamma detector are installed to prevent any potential for criticality. In addition, the accumulation rate of uranium in the Separations Building GEVS is very low compared with the safe mass of 12.2 kg U (26.9 lb U) assuming double batching and all the uranium were enriched to 6.0 $^{w}/_{o}$. These low accumulations coupled with the weight trip and gamma detectors render a criticality accident in the Separations Building GEVS highly unlikely.

3.4.9.2 Technical Services Building GEVS

The TSB GEVS provides exhaust of potentially hazardous contaminants. The system is shown on Figure 3.4-18, Process Flow Diagram Gaseous Effluent Vent System Technical Services Building, Sheets 1 and 2.

The GEVS servicing the TSB is located on the first floor of the TSB and is monitored from the Control Room.

3.4.9.2.1 Functional Description

Potentially contaminated exhaust air comes from the following rooms and services within the TSB:

Ventilated Room	2,700 m^3/hr	(1,589 cfm)
Laundry	1,000 m^3/hr	(589 cfm)
Fomblin Oil Recovery System	2,000 m^3/hr	(1,177 cfm)
Decontamination Workshop	12,300 m^3/hr	(7,240 cfm)
Chemical Laboratories	1,000 m^3/hr	(589 cfm)
Cylinder Preparation Room	1,000 m^3/hr	(589 cfm)
Solid Waste Collection Room	700 m^3/hr	(412 cfm)

Air from the Fomblin Oil Recovery System is part of the Decontamination Workshop discharge. Thus, the total airflow to be handled by the TSB GEVS is 18,700 m^3/hr (11,000 cfm). Flow rates and capacities are preliminary and are subject to change during final design.

The design requirements for the facility provide a large safety margin between normal and accident conditions so that no single failure could result in the release of significant hazardous material. The amounts of UF_6 in the system also preclude the release of significant quantities of hazardous material from a single failure or multiple failures. Instrumentation is provided to detect abnormal process conditions so that the process can be returned to normal by operator actions.

These requirements and operating conditions also assure "as low as reasonably achievable" personnel exposure to hazardous materials and compliance with environmental and safety criteria.

3.4.9.2.2 Major Components

The TSB GEVS consists of the following major components.

A. Duct system

B. Prefilter

C. HEPA filter

D. Impregnated carbon filter (impregnated with potassium carbonate)

E. Centrifugal Fan

F. Monitoring and controls (HF) before and after filters

G. Automatically controlled inlet and outlet isolation dampers

H. Exhaust stack

I. Gamma monitor and controls (prefilter and HEPA filter)

J. Monitoring and controls (alpha and HF) in exhaust stack

K. Stack Sampling system.

3.4.9.2.3 Design Description

The design bases and specifications are given in Table 3.4-16, Gaseous Effluent Vent System Design Bases (Technical Services Building).

The GEVS serving the TSB consists of a duct network that serves all of the uranium processing systems and operates at negative pressure. The ductwork is connected to one filter station and vents through one fan. Both the filter station and the fan can handle 100% of the effluent. There is no standby filter station or fan. Operations that require the GEVS to be operational are shut down if the system shuts down. The system capacity is estimated to be 18,700 m^3/hr (11,000 cfm). A differential pressure controller controls the fan speed and maintains negative pressure in front of the filter station.

Gases from the UF_6 processing systems pass through the 85% efficient prefilter which removes dust and protects the HEPA filter, then through the 99.97% efficient HEPA filter which removes uranium aerosols (mainly UO_2F_2 particles). Finally the air passes through the 99% efficient activated carbon (potassium carbonate impregnated) filter which captures HF. The remaining clean gases pass through the fan, which maintains the negative pressure upstream of the filter stations. The clean gases are then discharged through the exhaust stack on the TSB.

A minimum velocity of 12.7 m/s (2,500 ft/min) is maintained in the duct system in order to ensure that particulate contaminants are conveyed through the ductwork without settling. Each section of the duct system has an orifice plate to maintain a minimum air velocity. Each section also has a damper to balance the individual flows in the system. Flexible exhaust hoses have a capture velocity of 0.75 m/s (150 ft/min). Fume hoods shall have a capture velocity of 0.5 m/s (100 ft/min).

The TSB GEVS provides ventilation and hazardous contaminant removal for the TSB through ductwork, via hoods vented by booster fans to the technical services area, the chemical laboratory, and the vacuum pump rebuild workshop.

The materials of construction, corrosion allowances, and fabrication specifications for the equipment and ductwork used in the GEVS are compatible with UF_6 and HF and are noncombustible.

3.4.9.2.4 Design and Safety Features

The TSB GEVS is designed to protect plant personnel against uranium and HF exposure.

The TSB GEVS is designed to meet all applicable NRC requirements for public and plant personnel safety and effluent control and monitoring. The system design also complies with applicable standards of OSHA, EPA, and state and local agencies.

The system filters contaminated gases, and continuously monitoring exhaust gas flow to the atmosphere. HF monitors and alarms are installed upstream of the filtration systems and immediately upstream of the exhaust stack to avoid the release of hazardous materials to the environment. The alarms are monitored in the Separation Plant Control Room.

The TSB GEVS provides for continuous monitoring and periodic sampling of the gaseous effluent in the exhaust stack in accordance with the guidance in Regulatory Guide 4.16 (NRC, 1985).

Gamma monitors measure the build-up of ^{235}U on prefilters and HEPA filter. Upon detection of high-high gamma levels in the TSB GEVS filter, the TSB GEVS trips.

The unit is located in a dedicated room in the TSB with the GEVS for the Separation Plant. The filters are bag-in/bag-out. The frequency of filter replacement will be determined during the design phase and this section will be revised accordingly.

If the TSB GEVS stops operating, material within the duct will not be released into the building because each of the TSB GEVS connections has a P-trap to catch entrained material that could otherwise fall back into the building from the ductwork during system failure.

The design and in-place testing of the TSB GEVS will be consistent with the applicable guidance in Regulatory Guide 1.140 (NRC, 2001), ASME AG-1-1997 (ASME, 1997), and ASME N510-1989 (ASME, 1989). The system includes a potassium carbonate impregnated activated charcoal filter for HF removal. As such, the portions of Regulatory Guide 1.140 (NRC, 2001), ASME AG-1-1997 (ASME, 1997), and ASME N510-1989 (ASME, 1989), which address activated charcoal filters for radioiodine removal are not applicable. The prefilter efficiency (85%) is based on testing in accordance with ASME AG-1-1997 (ASME, 1997). The HEPA filter efficiency (99.97%) is based on removal of 0.3 micron particles when tested in accordance with ASME-AG-1 (ASME, 1997). The impregnated charcoal filter efficiency (99%) for removal of HF is based on Urenco specifications. In-place testing and inspections of the filters will be performed in accordance with the guidance in Regulatory Guidance 1.140 (NRC, 2001). The frequency for performance of in-place filter testing and the acceptance criteria for penetration and leakage (or bypass) will be consistent with the guidance in Regulatory Guide 1.140 (NRC, 2001). Qualification testing, to verify HF removal efficiency, of the impregnated charcoal will be performed using ASTM D6646-03 (ASTM, 2003), modified to reflect removal of HF instead of hydrogen sulfide. Laboratory testing of the impregnated charcoal filter of charcoal samples will be performed on an annual basis. Throughout the useful life of the impregnated charcoal, the impregnate is progressively consumed. The laboratory testing will determine the impregnant content within the sample. The amount of impregnant present in the sample is indicative of the remaining life of charcoal bed for removal of HF.

3.4.9.2.5 Instrumentation

The process variables, pressure, fan speed, and damper positioning are all controlled automatically. The fan speed is automatically controlled to maintain negative pressure in the system. The differential pressure across the filters is monitored and the fan speed is adjusted to maintain the design airflow rates. When a high pressure drop is detected across the filters, an alarm alerts the personnel that a filter change may be necessary. HF monitors measure the concentration of the gas in the air stream. Also, devices are used to measure the level of radiological contamination (alpha only) present in the air stream located in the stack. Deviations from specified values are indicated by alarms. HF and alpha monitors and alarms are installed upstream of the filtration system and immediately upstream of the exhaust stack to avoid the release of hazardous materials. The HF and radiological monitoring devices have non-interruptible power supplies in order to continue to function during a general power failure.

Each area has an alarm that is activated in the event that the TSB GEVS or the fan fails.

The TSB GEVS control system is mounted in a Local Control Center (LCC). This is a stand-alone system that does not generate alarms during normal operation. The LCC provides automatic control of the fan and dampers and provides local control via a Local Operator Interface (LOI) that is mounted in the LCC.

The Central Control System (CCS) has no supervisory control over the TSB GEVS control system. However, the TSB GEVS LCC communicates with the CCS via the dual redundant process network so that comprehensive monitoring of the TSB GEVS status exists. Data that is monitored is fan status, filter and duct pressure measurements, and damper status.

The TSB GEVS LCC has one PLC that provides all automatic control and protection required for the system and also the communication interface to the PCS. All equipment related to the TSB GEVS is directly wired to the LCC.

The radiological activity and HF monitoring instruments are stand-alone and powered separately. These instruments interface with the TSB GEVS LCC via hardwired signals that indicate when alarm limits have been exceeded.

Any shutdown device for the filter train and fan is latched and requires local operator action to reset.

High-level environmental alarms will shut down the TSB GEVS.

3.4.9.2.6 Criticality Safety

Within the TSB Ventilated Room, chemical traps will be emptied and product cylinders may be brought into the room for valve changes and subsequent testing. In the case of the traps there will be a mixture of product, feed and dump traps with a few from the tails operations. The product traps will be 10 kg (22.0 lb) carbon traps with a maximum holdup of 12 kg (26.5 lb) UF_6. The traps will have been de-gassed prior to being removed from the plant and there will be very little of the UF_6 absorbed on the trap that could become airborne. There may be a small amount of carbon drawn into the TSB GEVS as a result of emptying the traps. With approximately 20 carbon traps processed per year it is not considered credible that kilogram quantities of uranium would be drawn into the TSB GEVS, before filters were changed out.

A possible scenario for the acute accumulation of enriched uranium from the Ventilated Room exists from the valve testing operations. For this operation a cylinder is taken into the room and the valve is removed. A new valve is fitted to the cylinder and the cylinder is then pressure tested. This involves pressurizing the container with nitrogen then evacuating. For this operation the cylinder is connected to a portable rig, which in turn exhausts to the TSB GEVS. Since all pumps are lubricated with a UF_6 compatible oil there is the remote possibility that UF_6 could be pumped directly from the cylinder to the TSB GEVS. Weight and temperature trips on the carbon trap in this rig prevent this transfer from occurring.

Within the TSB Decontamination System there are a number of cleaning tanks. Components entering these tanks will have either been cleaned or de-gassed. It is not considered likely that significant quantities of uranium would enter the TSB GEVS as a result of these decontamination operations or the subsequent processing of the residues. The facility also provides the plant with a sample bottle cleaning service. Type 1S sample bottles delivered to the facility will be cleaned provided that there is no more than 20 g (0.04 lb) of residual material within the bottles. Even if this was all UF_6 and the bottle was opened the operator would see white hydrogen fluoride fume and there may be some small quantity of UF_6 associated with the release. Many mal-operations would be required for the TSB GEVS to see the quantity of material that would be needed to initiate a criticality.

Before pumps enter the TSB Contaminated Workshop there is a requirement for them to be de-gassed prior to transfer. It would be unusual for pumps to enter the facility with significant quantities of UF_6 remaining within the pump, including UF_6 dissolved in the Fomblin oil. On entering the facility the pumps are taken to the outgas area where the oil is removed. If dissolved UF_6 were present in the oil then there would be some fuming this would mainly be as a result of the dissolution of the UF_6 from the oil reacting with the water in the air. This would produce UO_2F_2 and HF. The HF would be drawn into the TSB GEVS and the majority of the UO_2F_2 would remain with the oil. The number of product pumps that cannot be successfully de-gassed is small and it is not considered that a significant fraction of the uranium in the oil would enter the TSB GEVS. Once the pumps have been transferred to the hydraulic table there will be uranium associated with the residual oil in the pump and some in the form of dry breakdown products. It is not considered possible that significant quantities of these will become airborne during the cleaning operations.

For the activities in the TSB, the accumulation rate of uranium in the TSB GEVS is very low compared with the safe mass of 12.2 kg U (26.5 lb U) assuming double batching and all the uranium were enriched to 6.0 $^{w}/_{o}$. These low accumulations coupled with regular sampling of filters, the weight trips and temperature trips, render a criticality accident highly unlikely.

3.4.10 Centrifuge Test and Centrifuge Post Mortem Processes

This section describes the basic components, functional requirements, and utilities required for operation of the Centrifuge Test Facility (CTF) and Centrifuge Post Mortem Facility (CPMF). The CTF and CPMF are located in the Centrifuge Assembly Building (CAB) as shown in Figure 3.3-13, Centrifuge Assembly Building, First Floor. These two facilities are segregated within the CAB for two reasons; the presence of uranium hexafluoride results in the areas being classified as process areas and the sensitive operations undertaken within the facilities require personnel access control. The functional requirements for the Centrifuge Test Facility and the Centrifuge Post Mortem Facility are presented in Table 3.4-17, Functional Requirements for Centrifuge Test and Post Mortem Facilities. Utility requirements for the two facilities are presented in Table 3.4-18, Utility Requirements for Centrifuge Test and Post Mortem Facilities.

3.4.10.1 Centrifuge Test Facility

3.4.10.1.1 Functional Description

The principal functions of the Centrifuge Test Facility (CTF) are to provide a means of functionally testing the performance of production centrifuges to ensure compliance with design parameters and to investigate production and operational problems. The facility consists of two test positions.

Testing in the CTF is performed by feeding a stream of gaseous UF_6 into the centrifuge and removing enriched and depleted streams, Product and Tails, respectively. During this process, the centrifuge is maintained at the required operating frequency, temperature, and pressure, and samples are taken from the Product and Tails streams to enable determination of the separative capacity of the centrifuge under test.

The discharge line from the mobile vacuum pump set and flexible exhaust hose is provided to the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System, see Section 3.4.10.3.

3.4.10.1.2 Major Components

The equipment located in the CTF comprises the following main components or sub-systems.

A. Centrifuge Cubicles

B. Centrifuge Inverter

C. Cooling Water System

D. UF_6 Feed and Take-off System

E. Chemical Trap and Vacuum Pump Sets

F. Supervisory Control and Data Acquisition System (SCADA)

G. Uninterruptible Power Supply (UPS)

H. Centrifuge Crash Detection System.

I. SCADA System.

J. Uninterruptible Power System (UPS).

K. Centrifuge Crash Detection System.

3.4.10.1.3 System Description

A. Centrifuge Cubicles.

The Centrifuge Cubicle consists of an insulated box manufactured from non-flammable insulating material. Each cubicle has front and top opening doors to facilitate access for loading and making process and utility connections.

A specially designed centrifuge mounting base plate and stand provides a solid mounting and attachment to the floor.

The test centrifuge is transported to a location immediately adjacent to the cubicle on a transport trolley. The centrifuge is then loaded into the cubicle using a jib crane with an electrically powered hoist. A platform is provided to make the process pipe work connections at the top of the centrifuge.

Air within the cubicle is maintained at a nominal operating set point, which is adjustable using an electrical heater located near the bottom of the cubicle, in conjunction with a circulating fan.

Cooling water is supplied through the wall of the Centrifuge Cubicle to the test centrifuge and subsequently returned to a local, dedicated Cooling Water System.

A flexible exhaust hose connected to the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System is positioned close to the centrifuge flange to provide local exhaust in the working area during disconnection from the facility. Appropriate gloves and positive pressure face mask with appropriate filtration is used during disconnection of any UF_6 process connections.

B. Centrifuge Inverter.

Each test position is provided with a variable speed inverter. The inverter provides a drive signal to the centrifuge motor. Drive up and drive down sequences are controlled by the SCADA system.

C. Cooling Water System.

The cooling water system is composed of a proprietary stand-alone unit. Heating and chilling capacity is required to enable delivery of a stable flow of water to both test positions. Supply and return connections are made to the test centrifuges mounted in the Centrifuge Cubicles.

D. UF_6 Feed and Take-off System.

The feed and take-off system consists of two identical stainless steel vessels; the UF_6 capacity of the system is 50 kg (110 lb).

Each vessel is fitted with cooling coils which carry liquid nitrogen to maintain the temperature at –70°C (-94°F) when used in take-off mode and heat tracing which maintains the temperature at

20°C (68°F) when used in feed mode. The neck of each vessel has heat tracing that is set to 25°C (77°F), irrespective of feed or take-off mode, preventing UF_6 desublimation in the inlet and outlet.

E. UF_6 Feed Supply.

Gaseous UF_6 is generated by a process of sublimation from one of the vessels, nominated the feed vessel. Energy required for sublimation is supplied by electrical heat tracing controlled to 20°C (68°F).

The feed is delivered from the feed vessel to the centrifuge, via a system of control valves and orifice plates, to achieve the required centrifuge feed pressure and flow rate.

F. UF_6 Take-off.

The enriched and depleted UF_6 streams are drawn from the centrifuge. Each stream is passed through an automatic control valve and orifice plate for flow measurement purposes. The streams are then merged and desublimed in the second vessel, nominated the take-off vessel. This vessel is chilled to –70°C (-94°F) using liquid nitrogen.

The piping/valve configuration allows each take-off stream to be diverted along an alternative route to allow a dedicated sample to be taken. A flexible tube connected to the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System is positioned close to the sample bottle during sample bottle connection and disconnection to provide local exhaust of the working area.

When all the UF_6 has been transferred to the take-off vessel, the previously heated feed vessel is cooled, and the previously cooled take-off vessel is heated, becoming the feed vessel, and allowing the UF_6 to be fed in the opposite direction.

The UF_6 can be recycled in this manner for approximately one year. A flexible tube connected to the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System is positioned close to the vessel during replacement of the UF_6 inventory to provide local exhaust of the working area.

G. CTF Feed and Take-off Vessel Recharging.

As stated previously, after approximately one year's operation it is necessary to replenish the system charge of about 50 kg (110 lb) UF_6.

This is affected by initially transferring the full UF_6 inventory into a single vessel. After this has been completed, the vessels are isolated and allowed to return to ambient temperature.

The process pipe work is evacuated and purged with nitrogen gas several times in a cyclic manner. Operational experience has shown that this procedure minimizes the possibility of UF_6 or HF release.

A flexible exhaust hose connected to the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System is positioned adjacent to the flange connection of the vessel isolation valve to provide local exhaust of the working area. The flange connection is then broken and blank flanges are fitted to the isolation valve and the facility process pipe work.

The vessel is emptied to an off-line UBC in the separation plant. The vessel is recharged from a feed cylinder and subsequently refitted to the centrifuge test facility.

H. Chemical Traps and Vacuum Pump Set.

The chemical traps and vacuum pump set are composed of a stainless steel trap filled with 10 kg (22.1 lb) of activated carbon, a stainless steel trap filled with 15 kg (33.1 lb) of aluminum oxide and a two stage rotary vane vacuum pump fitted with an nitrogen purge. The carbon trap of the chemical traps and vacuum pump set has a weighing system that will automatically trip the associated vacuum pump on high carbon trip weight.

The vacuum pump has upstream and downstream filters to prevent oil migration and discharges to the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System. These items are located on a movable skid.

The chemical traps and vacuum pump set provides the following functionality:

1. Initial evacuation of the test centrifuge.
2. Removal of UF_6 from the centrifuge and connecting pipe work during testing in the event the normal take-off route becomes unavailable.
3. Removal of non-condensable gases, which accumulate in the chilled take-off vessel during testing.
4. System purging at the end of testing; the centrifuge is evacuated and purged several times with nitrogen gas through a control valve which limits the rate of pressure change.

I. SCADA System

The centrifuge test facility has a dedicated control and data acquisition system. Control functions are performed using a Programmable Logic Controller (PLC). Independent hard wired trips are used for safety related functions.

The operator interfaces with the SCADA system via a computer terminal. The operator interface displays real time values and trends of all instruments associated with the centrifuge test facility and allows selection of various process modes and initiation of sequences.

J. Uninterruptible Power System (UPS).

A UPS is required to provide backup power to the PLC, the operator interface, and the hardwired safety circuits.

K. Centrifuge Crash Detection System.

Each test position is fitted with a centrifuge Crash Detection System. This system consists of a shock sensor, that is strapped to the test centrifuge, and signal processing electronics. The signal processor provides a digital input to the SCADA system PLC that, in turn, initiates a system shutdown and provides an alarm signal.

3.4.10.1.4 Design and Safety Features

As stated previously, control of the Centrifuge Test Facility is undertaken via the SCADA system. All process states and sequences are initiated by the operator. The operator can override any sequence and take manual control of the facility.

There are few hazards associated with the facility. The principal hazards are centrifuge failure or heat tracing failure of the feed vessel resulting in overheating of the vessel.

The safety enclosure for the centrifuge containment is well established and underpinned with experimental evidence.

In the event of an electrical heating or heat trace control failure, the design is such that with continuous maximum power input to the heating elements, no damage to the equipment can occur.

The electrical heating and heat tracing circuits of the UF_6 feed and take-off vessels are each fitted with two resistance temperature devices (RTDs). One RTD is used for control. The second RTD provides an independent fail-safe, hardwired trip of the heat tracing, set at 35°C (95°F). An independent capillary temperature sensor for automatic, fail safe, high temperature trip of the heat tracing is also provided. This value has been selected to prevent the formation of UF_6 gas at above atmospheric pressure.

The power to these electrical circuits is also removed if the pressure at the UF_6 feed or take-off vessel exit rises above 120 mbar (1.74 psia).

3.4.10.2 Centrifuge Post Mortem Facility

3.4.10.2.1 Functional Description

The principal functions of the Centrifuge Post Mortem Facility (CPMF) are as follows:

A. Facilitate dismantling of contaminated centrifuges using equipment and processes that minimize the potential to contaminate personnel or adjacent facilities.

B. Collect potentially contaminated components for transfer to the Solid Waste Collection Room in the TSB prior to disposal.

Operational experience to-date has shown that the demand for centrifuge post mortems is infrequent.

Centrifuges are brought into the CPMF from the cascade hall on a specially designed transport cart. The CPMF is used for careful, diligent dismantling of centrifuges. The centrifuges will have been operating in UF_6 and are therefore contaminated. The facility is equipped with radiological monitoring devices (alpha in air), toilets and washing facilities, and hand, foot, and clothing personnel monitors to detect surface contamination. Wash water is collected and monitored for contamination prior to discharge. All ventilation exhausts are routed through the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System. Flexible exhaust hoses, that are connected to the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System, are positioned by the operator local to the centrifuge prior to commencing the dismantling process.

Atmospheric conditions within these two facilities require control. To facilitate this requirement, an airlock entry is employed. For additional functional and utility requirements see Table 3.4-17, Functional Requirements for Centrifuge Test and Post Mortem Facilities, and Table 3.4-18, Utility Requirements for Centrifuge Test and Post Mortem Facilities.

3.4.10.2.2 Major Components

The equipment located in the Centrifuge Post Mortem Facility consist of the following main components or sub-systems:

A. Centrifuge dismantling facility

B. Centrifuge manipulation equipment

C. Inspection facilities

D. Solid and liquid waste collection and segregation facilities.

3.4.10.2.3 System Description

A. CPMF Centrifuge Dismantling Facility.

The centrifuge dismantling facility is composed of a stand, onto which the centrifuge is mounted, a local jib crane, and miscellaneous tools.

The stand has an elevated working platform to allow access to the top of the centrifuge. The platform is large enough to accommodate two people, necessary tools to enable dismantling, and a lay down area for potentially contaminated components.

A jib crane is located over the stand to enable centrifuge removal from and replacement to the transport cart, and to facilitate loading and unloading the stand.

Miscellaneous tools are used to dismantle the centrifuge. These tools are solely for the purpose of centrifuge post mortem and are stored adjacent to the dismantling facility.

A flexible exhaust hose from the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System is positioned adjacent to the centrifuge enclosure to provide local exhaust in the working area during dismantling.

The dismantling facility has to deal with both intact and crashed centrifuges. The dismantling processes are consequently different.

Dismantling of intact centrifuges is relatively easy. Removal of the internals is facilitated by use of the jib crane.

Crashed centrifuges, however, yield fragmented debris. To contain the spread of potentially contaminated debris, a dedicated vacuum cleaner is used to capture particulates. The dedicated vacuum cleaner complies with the requirement to be safe by shape to prevent the possibility of criticality. Removal of the internals often requires inversion of the centrifuge casing to retrieve component parts for subsequent inspection. This operation is undertaken using the centrifuge manipulation equipment.

Operational restrictions are placed on personnel undertaking post mortem activities. These are summarized as follows:

All personnel must utilize personal protection equipment that is identified via a risk assessment and follow operational procedures to undertake post mortem activities.

To minimize potential for criticality, only one centrifuge at a time can be dismantled within the facility. Aqueous and non-aqueous cleaning agents are not allowed in the centrifuge post mortem facility. Component cleaning can only be carried out using dry wipe techniques.

B. Centrifuge Manipulation Equipment.

The centrifuge manipulation equipment is a piece of mechanical handling equipment that provides for rotation of the centrifuge casing.

C. Inspection Facilities.

An inspection area is located within the centrifuge post mortem facility to facilitate collection of evidence to support failure hypotheses. The inspection facilities have an inspection bench, portable lighting, a microscope, an endoscope, and a digital video camera.

D. Solid and Liquid Waste Collection and Segregation Facilities.

Waste from centrifuge post mortem consists of small quantities of both non-aqueous liquid and dry solids.

The non-aqueous liquid waste is transferred into a 5 L (1.32 gal) plastic container. This container is stored in the centrifuge post mortem facility until it is full. The full container is subsequently transferred to the Solid Waste Collection Room in the TSB. It is then characterized, packaged, and sent for disposal.

The solid wastes are segregated into like materials prior to disposal. Some of the items are required to be broken down to reduce volume and ease handling. This is carried out using a mechanical bench saw. Wastes are then bagged and monitored to determine the level of surface contamination. The containerized wastes are sent to the Solid Waste Collection Room in the TSB for disposal.

3.4.10.2.4 Design and Safety Features

Historical operational experience in Europe has shown that centrifuge post mortems are infrequent events. It is envisioned that no post mortem activity is required during early operational life. Consequently, it is expected that no more than 20 post mortems would be undertaken over the life of the facility.

Waste material such as carbon fiber, metal (principally aluminum), oil, paper, wipes, gloves, and contaminated disposable clothing is generated. Operational experience in Europe has shown that uranium is found as surface contamination in the form of either UO_2F_2 or uranium tetrafluoride (UF_4).

3.4.10.3 Centrifuge Test and Post Mortem Facilities Exhaust Filtration System

The Centrifuge Test and Post Mortem Facilities Exhaust Filtration System provides exhaust of potentially hazardous contaminants from the Centrifuge Test and Post Mortem Facilities. The system also ensures the Centrifuge Post Mortem Facility is maintained at a negative pressure with respect to adjacent areas. The system is shown on Figure 3.4-19, Process Flow Diagram Centrifuge Test and Post Mortem Facilities Exhaust Filtration System.

The Centrifuge Test and Post Mortem Facilities Exhaust Filtration System is located in the Centrifuge Assembly Building and is monitored from the Control Room.

3.4.10.3.1 Functional Description

Potentially contaminated exhaust air comes from the Centrifuge Test and Post Mortem Facilities. The total airflow to be handled by the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System is 9,345 m^3/hr (5,500 cfm). Flow rates and capacities are preliminary and are subject to change during final design.

The design requirements for the facility provide a large safety margin between normal and accident conditions so that no single failure could result in the release of significant hazardous material. The amounts of UF_6 in the system also preclude the release of significant quantities of hazardous material from a single failure or multiple failures. Instrumentation is provided to detect abnormal process conditions so that the process can be returned to normal by operator actions.

These requirements and operating conditions also assure "as low as reasonably achievable" personnel exposure to hazardous materials and compliance with environmental and safety criteria.

3.4.10.3.2 Major Components

The Centrifuge Test and Post Mortem Facilities Exhaust Filtration System consists of the following major components.

- Duct system
- Prefilter
- Impregnated carbon filter (impregnated with potassium carbonate)
- High Efficiency Particulate Air Filter (HEPA)
- Two exhaust filtration fans
- Exhaust stack
- Stack alpha monitor
- Stack HF monitor.

3.4.10.3.3 Design Description

The Centrifuge Test and Post Mortem Facilities Exhaust Filtration System consists of a duct network that serves the Centrifuge Test and Post Mortem Facilities and operates at negative pressure. The ductwork is connected to one filter station and vents through either of two 100% fans. Both the filter station and either of the fans can handle 100% of the effluent. One of the fans will normally be in standby. Operations that require the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System to be operational are manually shut down if the system shuts down. The system capacity is estimated to be 9,345 m^3/hr (5,500 cfm).

Gases from the associated areas pass through the 85% efficient prefilter which removes dust and protects the carbon filter, then through the 99% efficient activated carbon (potassium carbonate impregnated) filter that captures HF. Remaining uranic particles (mainly UO_2F_2 particles) will be filtered by the 99.97% efficient HEPA filter. The remaining clean gases pass through a fan, which maintains the negative pressure upstream of the filter station. The clean gases are then discharged through the stack on the Centrifuge Assembly Building.

A minimum velocity is maintained in the duct system in order to ensure that particulate contaminates are conveyed through the ductwork without settling. Each section also has a damper to balance the individual flows in the system. Flexible exhaust hoses are provided in both the Centrifuge Test Facility and the Centrifuge Post Mortem Facility. A hood is also provided in the Centrifuge Post Mortem Facility.

The materials of construction, corrosion allowances, and fabrication specifications for the equipment and ductwork used in the GEVS are compatible with UF_6 and HF and are noncombustible.

3.4.10.3.4 Design and Safety Features

The Centrifuge Test and Post Mortem Facilities Exhaust Filtration System is designed to protect plant personnel against uranium and HF exposure.

The Centrifuge Test and Post Mortem Facilities Exhaust Filtration System is designed to meet all applicable NRC requirements for public and plant personnel safety and effluent control and monitoring. The system design also complies with applicable standards of OSHA, EPA, and state and local agencies.

The Centrifuge Test and Post Mortem Facilities Exhaust Filtration System provides for continuous monitoring and periodic sampling of the gaseous effluent in the exhaust stack in accordance with the guidance in Regulatory Guide 4.16 (NRC, 1985).

The system filters contaminated gases, and continuously monitoring exhaust gas flow to the atmosphere. The system also provides primary confinement for the Centrifuge Post Mortem Facility by maintaining the Centrifuge Post Mortem Facility at a negative pressure relative to adjacent areas. An HF monitor and associated alarm and an alpha radiation monitor and associated alarm are installed immediately upstream of the exhaust stack to avoid the release of hazardous materials to the environment. The frequency of filter replacement will be determined during the design phase and this section will be revised accordingly.

The design and in-place testing of the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System will be consistent with the applicable guidance in Regulatory Guide 1.140 (NRC, 2001), ASME AG-1-1997 (ASME, 1997), and ASME N510-1989 (ASME, 1989). The system includes a potassium carbonate impregnated activated charcoal filter for HF removal. As such, the portions of Regulatory Guide 1.140 (NRC, 2001), ASME AG-1-1997 (ASME, 1997), and ASME N510-1989 (ASME, 1989), which address activated charcoal filters for radioiodine removal are not applicable. The prefilter efficiency (85%) is based on testing in accordance with ASME AG-1-1997 (ASME, 1997). The HEPA filter efficiency (99.97%) is based on removal of 0.3 micron particles when tested in accordance with ASME-AG-1 (ASME, 1997). The impregnated charcoal filter efficiency (99%) for removal of HF is based on Urenco specifications. In-place testing and inspections of the filters will be performed in accordance with the guidance in Regulatory Guidance 1.140 (NRC, 2001). The frequency for performance of in-place filter testing and the acceptance criteria for penetration and leakage (or bypass) will be consistent with the guidance in Regulatory Guide 1.140 (NRC, 2001). Qualification testing, to verify HF removal efficiency, of the impregnated charcoal will be performed using ASTM D6646-03 (ASTM, 2003), modified to reflect removal of HF instead of hydrogen sulfide. Laboratory testing of the impregnated charcoal filter of charcoal samples will be performed on an annual basis. Throughout the useful life of the impregnated charcoal, the impregnate is progressively consumed. The laboratory testing will determine the impregnant content within the sample. The amount of impregnant present in the sample is indicative of the remaining life of charcoal bed for removal of HF.

3.4.10.3.5 Instrumentation

The process variables, pressure, fan speed, and damper positioning are all controlled automatically. The fan speed is automatically controlled to maintain negative pressure in the system. The differential pressure across the filters is monitored to provide indication of when filter replacement is required. An HF monitor measures the concentration of the gas in the air stream. Also, a radiation detector is used to measure the level of radiological contamination (alpha only) present in the air stream located in the stack. Deviations from specified values for HF and alpha radiation are indicated by alarms. The HF and alpha radiation monitoring devices have non-interruptible power supplies in order to continue to function during a general power failure.

3.4.11 Material Handling Processes

The NRC staff previously reviewed the Claiborne Enrichment Center SAR application relative to the Material Handling Processes and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion on the Material Handling Processes is provided in NUREG-1491 (NRC, 1994), Sections 3.1 and 3.2.

The NRC in Bulletin 2003-03 (NRC, 2003), Potentially Defective 1-in valves for Uranium Hexafluoride Cylinders, identified performance and safety concerns with 1-in valves for UF_6 cylinders manufactured by the Hunt Valve Company. In response to Bulletin 2003-03 (NRC, 2003), LES will not purchase UF_6 cylinders with the 1-in Hunt valves installed nor purchase any replacement 1-in valves from Hunt.

In the unlikely event that any cylinders are received at the NEF with the 1-in Hunt valves installed, the following actions will be taken.

- If the cylinder is empty, the valve will be replaced before the cylinder is used in the facility.
- If the cylinder is filled, a safety justification to support continued use of the cylinder until the valve can be replaced will be developed or the valve will be replaced in accordance with NEF procedures.

No cylinders with the 1-in Hunt valve installed will be used as UBCs.

3.4.11.1 Cylinder Receipt and Shipping

The Cylinder Receipt and Dispatch Building (CRDB) provides for handling of feed cylinders, product cylinders, semi-finished product cylinders, prepared empty cylinders and UBCs, and provides space for the following services:

- Cylinder loading and unloading
- Inventory weighing
- Secure internal storage (no UBC or empty feed storage in CRDB)

- Preparation and storage area for overpack/protective structural packaging.

The cylinders are received, shipped offsite, stored, and transferred to and from the UF_6 Handling Areas, Blending and Liquid Sampling Area, and UBC Storage Pad.

Prepared empty cylinders, semi-finished product cylinders, full feed cylinders, and final product cylinders are stored in the CRDB.

Full UBCs and empty feed cylinders are not stored in the CRDB. They are transported through the TSB and stored in the UBC Storage Pad.

The CRDB layout is shown on Figure 3.3-10, Cylinder Receipt and Dispatch Building, First Floor, Part A, and Figure 3.3-11, Cylinder Receipt and Dispatch Building, First Floor, Part B. The UF_6 Feed cylinder delivery and storage requirements are presented in Table 3.4-19, UF_6 Feed Cylinder Delivery and Storage Requirements.

3.4.11.1.1 Description

The majority of the floor area in the CRDB is used as a storage or staging area for feed and product cylinders. The cylinders are placed on concrete saddles to stabilize them while they are stored in this area. Different size saddles are provided for 48-in and 30-in cylinders. The cylinders are positioned such that access is possible from an overhead crane.

Trucks arrive at the building carrying feed cylinders, empty UBC or product cylinders, and enter through the main vehicle loading bay. This bay is equipped with vehicle access platforms that aid with cylinder loading and unloading operations.

Unloaded trucks either leave the site or remain in a staging area adjacent to the CRDB. Trucks in this staging area await cylinders that are to be shipped from the site.

3.4.11.1.2 Equipment

The following equipment is used for cylinder handling in the CRDB.

A. Vehicle Loading And Unloading Platform.

The vehicle loading and unloading platforms are located adjacent to the main transport vehicle access doorways. These platforms provide a safe method of transfer to the vehicle trailer while loading and unloading activities are in progress. Cylinders will be stored a minimum of one meter from the vehicle platform to eliminate the fire hazard associated with trucks in the CRDB.

B. Double Girder Bridge Cranes.

Two double girder bridge cranes handle the cylinders in the CRDB. The cranes span half the width and run the full length of the main storage building. They are operated by an automated control system and equipped with remotely operated grabs. Each hoist has a maximum lift of 9 m (29.5 ft). Crane movement requirements are presented in Table 3.4-20, Crane Movement Requirements. The minimum lift is based upon the following data:

- Floor to top height of a vehicle mounted ISO container 4.1 m (13.4 ft)
- Lift clearance between ISO container and underside of cylinder 0.6 m (2 ft)
- Allowance for a 48 in cylinder 1.2 m (3.9 ft)

• Typical length of a universal cylinder grab (including fixing)	2.0 m (6.6 ft)
• Allowance for unknown effect of a 48-in cylinder overpack	1.0 m (3.25 ft)
• Total	8.9 m (29.16 ft)

The crane specifications are as follows:

• Span	20 m	(65.6 ft)
• Capacity	20 MT	(44,100 lb)
• Hoist lift height	9 m	(29.5 ft)
• Hoist lift speed (Variable Frequency Drive (VFD))	6 m/min	(20 ft/min)
• Travel length	225 m	(708.67 ft)
• Bridge travel speed (VFD)	49 m/min	(161 ft/min)
• Brake type	Direct Current Disc	

ISO containers are International Organization for Standardization Series 1 freight containers that are supplied in accordance with the ISO 668:1995 (ISO, 1995) Standard. These containers are used for intercontinental shipping. They are 2,438 mm (8 ft) wide and are available in a variety of heights ranging from 2,438 mm (8 ft) to 2,896 mm (9.5 ft).

C. Scales.

Each cylinder that enters or exits the CRDB is weighed. Weigh scales capable of weighing a load of 17 MT (37,500 lb) and capable of accepting a load of 20 MT (44,100 lb) are required on each end of the CRDB. One set of scales is utilized in the area adjacent to the cylinder truck loading/unloading bay. The other set of scales is located in the area adjoining the Blending and Liquid Sampling Area. The scales are capable of weighing to a tolerance of ±2.5 kg (±5.5 lb). The scales have a reader and printout facilities, and are located in a pit such that the weigh table is flush with the finished building floor slab.

D. Flatbed Trucks And Rail Transporters.

After processing, the cylinders are transported between the CRDB, the UF_6 Handling Areas, and the UBC Storage Pad via flatbed trucks. A double girder Gantry crane is used to manage the cylinders in the UBC Storage Pad.

3.4.11.1.3 Cylinder Specifications

Cylinders stored and handled in the CRDB vary in size and weight from 30B cylinders to 48Y cylinders. The cylinders have the following characteristics:

<u>30B Cylinder</u>		
Weight of UF_6	2,277 kg	(5,020 lbs)
Gross cylinder weight	2,912 kg	(6,420 lbs)
Diameter	762 mm	(2.5 ft)
Length	2,070 mm	(6.8 ft)

48Y Cylinder

Weight of UF_6	12,501 kg	(27,560 lbs)
Gross cylinder weight	14,860 kg	(32,761 lbs)
Diameter	1,232 mm	(4.08 ft)
Length	3,728 mm	(12.25 ft)

48X Cylinder

Weight of UF_6	9,539 kg	(21,030 lbs)
Gross cylinder weight	11,580 kg	(25,530 lbs)
Diameter	1,220 mm	(4 ft)
Length	3,020 mm	(9.9 ft)

3.4.11.1.4 CRDB Storage Areas

The CRDB accommodates the following areas:

Final product storage	330 m^2	(3,552 ft^2)
Overpack storage (72 overpacks)	440m^2	(4,736 ft^2)

3.4.11.1.5 Product Cylinder Storage

Semi-finished product cylinder storage areas are shown on Figure 3.3-10, Cylinder Receipt and Dispatch Building, First Floor, Part A, and final product storage areas are shown on Figure 3.3-11, Cylinder Receipt and Dispatch Building, First Floor, Part B. The areas accommodate 125 semi-finished cylinders and 125 final product cylinders.

Site vehicle access/single loading bay	400 m^2	(4,306 ft^2)
Full feed cylinder storage	6,231 m^2	(67,070 ft^2)
Prepared (empty) cylinder storage	400 m^2	(4,306 ft^2)
Semi-finished product storage	330 m^2	(3,552 ft^2)
Preparation Area	400 m^2	(4,306 ft^2)

3.4.11.1.6 Feed Cylinder Storage

Feed cylinder storage areas are shown on Figure 3.3-10 and on Figure 3.3-11. Feed material is stored under vacuum in corrosion resistant Type 48Y or 48X cylinders. The CRDB provides enough space to store up to 708 cylinders. These cylinders can be stored without providing room for cylinder maintenance because they are only in temporary storage. Based on this type of design, the area allocated per feed cylinder is 8 m^2 (86 ft^2). Thus, the maximum storage area required is 5664 m^2 (60,967 ft^2). A 10% allowance is reserved for staging purposes, bringing the total required area to 6,231 m^2 (67,070 ft^2).

3.4.11.1.7 Cylinder Deliveries

Cylinder deliveries to and from the site generally consist of feed deliveries to the site, product transport from the site, and return of supplier empty feed cylinders. At the NEF, full 48X cylinders are delivered one cylinder per delivery vehicle. Full 48X cylinders may be delivered two cylinders per delivery vehicle. New empty 48-in cylinders are delivered nine cylinders per delivery vehicle. Empty washed out 48-in cylinders are delivered six cylinders per vehicle. The

30-in product cylinders are delivered four cylinders per 6 m (20 ft) of delivery vehicle. The number of product cylinders per vehicle can vary and a typical shipment frequency would be one vehicle per 3 days (122 shipments per year). This information for a total plant capacity of 3 million SWU per year is summarized below. The figures in the following table represent a maximum number of deliveries per year. An alternate cylinder management strategy whereby empty feed cylinders are refilled with tails and new empty 48Y cylinders are provided to the feed suppliers would reduce the number of NEF deliveries.

Delivery Description	Number cylinders per year	Number cylinders per vehicle	Number deliveries per year
Feed In	690	1	690
Empty Tails In	625	9	70
Product Out	350	4	88
Empty Feed Out	690	6	115
Total	-	-	963

3.4.11.2 Cylinder Transport within the Facility

3.4.11.2.1 Cylinder Transport Between CRDB and the Product Blending and Liquid Sampling Area

Two double girder bridge cranes in the CRDB are used to move cylinders to either of the two weighing stations at the end of the CRDB. Cylinders moving from the CRDB to the Blending and Liquid Sampling Area and vise versa may be weighed. Each of the weighing stations has a transporter to convey the cylinders from the CRDB to the Blending and Liquid Sampling Area. The transporters travel along rails embedded in the floor. At rail intersections, physical stops prevent the CRDB transporter from colliding with the UF_6 Handling Area transporter. The rail system is depicted on Figure 3.3-10, Cylinder Receipt and Dispatch Building, First Floor, Part A.

A total of two rail transporters for the CRDB to UF_6 Handling/Blending and Liquid Sampling are included in the facility. The transporters may be battery powered, or fed by an electric feeder.

Cylinders are empty product, product, empty feed, feed, empty UBCs, UBCs, or semi-finished product cylinders.

3.4.11.2.2 Cylinder Transport Between the Product Blending and Liquid Sampling Area and the TSB

Cylinders are transported between the Blending and Liquid Sampling/ UF_6 Handling transporter and the TSB by a rail transport device that travels along rails embedded in the floor. Once the cylinders are in the TSB, they are lifted and moved with a bridge crane hoist system located in the Cylinder Preparation Room.

One rail transporter between the UF_6 Handling/Blending and Liquid Sampling and the TSB is installed in the facility. The transporter may be battery powered, or fed by an electric feeder.

New or clean cylinders are empty product, empty feed or empty tails. See Section 3.3.1.2.2.5 for details of cylinder preparation.

3.4.11.2.3 Cylinder Testing

When cylinders are delivered without valves and plugs, an internal inspection of the washed out or new cylinders is made in the Cylinder Preparation Room using a conventional remote optical viewing device, called an Endoscope. 48-in cylinders that are supplied with fitted valves and plugs do not require testing. All 30-in cylinders are inspected internally for criticality safety purposes.

Cylinders are pressure tested using compressed air in accordance with ANSI N14-2001 (ANSI, 2001). This system is used for testing new and decontaminated empty cylinders only. The test procedure is automated and is performed after the valve and plug fitting activities have been completed. The pressure test is administered via a set of program controlled automatic valves.

3.4.11.2.4 Cylinder Transport Between the Product Blending and Liquid Sampling Area and the UF_6 Handling Areas

A rail system extends between the Blending and Liquid Sampling Area and all of the UF_6 Handling Areas. The rail has two independent rail transporters. Each of the transporters has a drawbridge that links the transporter to the appropriate station or adjoining transporter. The UF_6 rail transporters are depicted in Figure 3.4-20, Rail Transporter Area Equipment Drawing. Its function is the transfer of cylinders to the appropriate Product Blending System Donor Station, Product Blending System Receiver Station, Product Liquid Sampling Autoclave, Solid Feed Station, Product Low Temperature Take-off Station, Tails Low Temperature Take-off Station or Feed Purification Low Temperature Take-off Station.

Cylinders are empty product, product, empty feed, feed, empty UBCs, UBCs or semi-finished product cylinders. Each of the transporters may be battery powered or fed by an electric feeder embedded in the concrete.

3.4.11.3 UBC Storage Pad

The NEF utilizes an area outside of the Cylinder Receipt and Dispatch Building (CRDB) for storage of UBCs. The UBC Storage Pad is used for storage of cylinders containing UF_6 that is depleted in ^{235}U. It is also used for the storage of empty feed cylinders. Access to the cylinder storage pad is controlled and a fence is provided so that only authorized vehicles may enter the area. The tails storage requirements are presented in Table 3.4-21, UBC Storage System Requirements.

3.4.11.3.1 Description

Space is allocated to provide storage of UBCs for 30 years of output from the facility. The uranium byproduct material is stored under vacuum in corrosion resistant Type 48Y cylinders. Empty feed cylinders are also Type 48Y cylinders.

The UBC Storage pad can accommodate storage of up to 15,727 48Y cylinders. The cylinders are stacked two high. Concrete saddles are used to store the cylinders approximately 200 mm (8 in) above ground level.

3.4.11.3.2 Equipment

The UBC Storage Pad layout is based on moving the cylinders with cranes and either diesel or electric flatbed trucks. Two double girder bridge cranes are used to load the depleted UF_6 cylinders onto the flatbed trucks in the CRDB. The trucks transport the cylinders from the CRDB to the double girder Gantry crane in the UBC Storage Pad. The Gantry crane is used to remove the cylinders from the flatbed trucks and place them on the UBC Storage Pad. The Gantry crane is designed to double stack the cylinders.

The specifications for the double girder Gantry crane are as follows:

Span	43.6 m (143 ft)
Capacity	20 MT (44,100 lb)
Hoist lift height (maximum)	9 m (30 ft)
Hoist lift speed (VFD)	6 m/min (20 ft/min)
Travel length	641 m (2,100 ft)
Bridge travel speed (VFD)	49 m/min (160 ft/min)
Trolley travel speed (VFD)	24 m/min (80 ft/min)
Brake type	Direct Current Disc

3.4.11.3.3 UBC Storage

The selected storage option is a double-stacked cylinder storage using a Gantry crane and flatbed trucks for cylinder handling. This type of storage arrangement facilitates visual inspection and removal of the cylinders for maintenance.

The total area for UBC storage for facility operation is approximately 8.5 ha (21 acres). These areas include a 10% allowance for staging activities, but do not include allocated areas for access or perimeter roads.

3.4.11.3.4 Empty Feed Cylinder Storage

Empty feed cylinders require a radiological cooling period in storage prior to return to the customer. The cooling period is dependent upon the emitted dose, and is typically three months. No additional spacing is required for gamma reading purposes. The area allocated per empty feed cylinder is 8 m^2 (86 ft^2). An allowance has been made for six months of storage of empty feed cylinders. This requires a space large enough to accommodate 354 cylinders, a total of 2832 m^2 (30,483 ft^2). With the 10% allowance for staging purposes, a total area of 3,115 m^2 (33,530 ft^2) is required. The area allocated for empty feed cylinders is located in the UBC Storage Pad.

3.4.12 References

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